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Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

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Annual 10 CFR 50.59 Summary Report for 2005

Nuclear Management Company, LLC (NMC), is submitting this annual 10 CFR 50.59 Summary Report for the Point Beach Nuclear Plant (PBNP).

This report consists of two enclosures. Enclosure 1 contains descriptions of facility changes, tests and experiments in accordance with 10 CFR 50.59 during 2005. Enclosure 2 contains commitment change evaluations completed in 2005.

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Enclosures (2)

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ENCLOSURE 1

POINT BEACH NUCLEAR PLANT
ANNUAL 10 CFR 50.59 SUMMARY REPORT FOR 2005

Modifications
FSAR Changes and Other Evaluations

MR 03-056, Replacement of Unit 2 Reactor Vessel Closure Head

Activity Description: Modification MR 03-056 replaced the Unit 2 Reactor Vessel Closure Head (RVCH). The replacement RVCH was furnished with new Control Rod Drive Mechanisms (CRDMs), new CRDM latch mechanism actuating coils, new thermal sleeve assemblies, a new penetration design for the core exit thermocouple instrument cables, a new lifting lug design, new piping for the reactor head vent and reactor vessel level instrumentation system (RVLIS), a new shroud support ring, new O-ring retaining clips, and new CRDM dummy cans. Various material and configuration changes were incorporated into the design of the replacement components to reduce their susceptibility to stress corrosion cracking and to reduce potential leakage paths. The reactor vessel lifting lugs and shroud support ring were redesigned to satisfy the lifting and loading requirement of the Head Assembly Upgrade Package (HAUP). The O-ring retaining clips were redesigned to ensure that the O-ring remains in its groove when the RVCH is removed from the reactor vessel. The CRDM dummy cans were redesigned such that they were mounted onto an adjacent CRDM rather than from the head adapter plug of a spare head penetration.

Summary of 10 CFR 50.59 Evaluation: The components installed by MR 03-056 were conservatively designed using the applicable design criteria and design codes. Their original design bases were maintained and all applicable design loads were considered. The designs ensure that the components will maintain their integrity and required design margins when subjected to the applicable normal, accident and design basis earthquake loading conditions. The material changes were evaluated and found to be acceptable for their intended service. MR 03-056 had no adverse impact on the performance of any existing plant system or component. MR 03-056 did not create conditions that are more limiting than those assumed in the current safety analyses. The design basis limits for reactor coolant system (RCS) pressure and stress were not exceeded or altered.

MR 03-056 did not result in an increase in the frequency of occurrence of a large break LOCA, small break LOCA, or control rod ejection accident. MR 03-056 did not result in an increase in the consequences of an accident previously evaluated in the licensing basis, nor did it create the possibility of an accident of a different type or a failure of a different type than any previously evaluated in the licensing basis. The design basis limits for the RCS (pressure and stress) were not exceeded or altered. No method of evaluation was affected. Pursuant to Paragraph (c)(1) of 10 CFR 50.59, the modification did not require a license amendment. (EVAL 2004-006)

MR 03-047, Replacement of Unit 1 Reactor Vessel Closure Head

Activity Description: Modification MR 03-047 replaced the Unit 1 Reactor Vessel Closure Head (RVCH). The replacement RVCH was furnished with new Control Rod Drive Mechanisms (CRDMs), new CRDM latch mechanism actuating coils, new thermal sleeve assemblies, a new penetration design for the core exit thermocouple instrument cables, a new lifting lug design, new piping for the reactor head vent and reactor vessel level instrumentation system (RVLIS), a new shroud support ring, new O-ring retaining clips, and new CRDM dummy cans. Various material and configuration changes were incorporated into the design of the replacement components to reduce their susceptibility to stress corrosion cracking and to reduce potential leakage paths. The reactor vessel lifting lugs and shroud support ring were redesigned to satisfy the lifting and loading requirement of the Head Assembly Upgrade Package (HAUP). The O-ring retaining clips were redesigned to ensure that the O-ring remains in its groove when the RVCH is removed from the reactor vessel. The CRDM dummy cans were redesigned such that they were mounted onto an adjacent CRDM rather than from the head adapter plug of a spare head penetration.

Summary of 10 CFR 50.59 Evaluation: The components installed by MR 03-047 were conservatively designed using the applicable design criteria and design codes. Their original design bases were maintained and all applicable design loads were considered. The designs ensured that the components will maintain their integrity and required design margins when subjected to the applicable normal, accident and design basis earthquake loading conditions. The material changes were evaluated and found to be acceptable for their intended service. MR 03-047 had no adverse impact on the performance of any existing plant system or component. MR 03-047 did not create conditions that were more limiting than those assumed in the current safety analyses. The design basis limits for reactor coolant system (RCS) pressure and stress were not exceeded or altered.

MR 03-047 did not result in an increase in the frequency of occurrence of a large break LOCA, small break LOCA, or control rod ejection accident. MR 03-047 did not result in an increase in the consequences of an accident previously evaluated in the licensing basis, nor did it create the possibility of an accident of a different type or a failure of a different type than any previously evaluated in the licensing basis. The design basis limits for the RCS (pressure and stress) were not exceeded or altered. No method of evaluation was affected. Pursuant to Paragraph (c)(1) of 10 CFR 50.59, the modification did not require a license amendment. (EVAL 2005-002)

While MR 03-056 (Unit 2) and 03-047 (Unit 1) did not require NRC approval prior to implementation, during April of 2005, it was identified that the current license basis relating to a RVH drop accident had not been updated to reflect correspondence placed on the plant docket in 1982. License Amendments 220 and 226 authorized changes to the design basis and FSAR related to a postulated reactor vessel head drop accident. The FSAR has been updated to reflect the revised design basis authorized by the amendments in accordance with 10 CFR 50.71(e). The description of a postulated reactor vessel head drop accident incorporated into the FSAR includes discussion of

occurrences that lead to the initiating event; the event frequency classification; the sequence of events from initiation of the final stabilized condition; plant characteristics considered in the safety evaluation; assumed protective system actions; core and system performance; barrier performance; and radiological consequences.

MR 99-035 and MR 99-036, Containment Hatch Airlock Equalizing Valve Replacement

Activity Description: MR 99-035*A/*B and MR 99-036*A/*B replaced the equalizing valves for the Equipment Access Airlock (C-1) and Personnel Access Airlock (C-2) in both units. The previous valves (C-1-D1V, C-1-D2V, C-2-D1V, C-2-D2V) were 2" ball valves. Each hatch door had a valve that vents the hatch to containment (inner door/D2V valves) or vents the hatch to atmosphere (outer door/D1V valves). The previous ball valves were very difficult to operate and had resulted in personal injury due to the excessive force required to operate the valves. The replacement "valves" are devices of a fundamentally different design. The new equalizing valves consist of a disc plate that slides along four rails, and is forced into a base plate by a linkage arm. The flange joint has a double O-ring seal configuration, with the test ports that allow pressure testing between the rings.

Summary of 10 CFR 50.59 Evaluation: The new equalization valves perform a function to isolate containment from the environment. They do not increase the probability of occurrence of an accident or create the possibility of a new accident since they only serve to isolate containment, which is a fission product barrier, and are opened to allow access to containment. The new valves do not increase the likelihood of a malfunction of equipment important to safety or increase the radiological consequences of an accident or malfunction. No departure from a method of evaluation exists for this change. A design basis limit for a fission product barrier was not exceeded or challenged. Pursuant to paragraph (c)(1) of 10 CFR 50.59, this change did not require a license amendment. (EVAL 2005-003)

Procedure Changes, Revisions to Procedures to Prevent Exceeding 13.8 kV System Current Interrupt Ratings during Gas Turbine Generator Load Tests

Activity Description: Revisions to procedures were made to address the potential of exceeding current interrupt ratings of 13.8 kV circuit breakers when the Gas Turbine Generator (G-05) is in operation. The following procedures were revised due to the potential to exceed current interrupt ratings of 13.8 kV circuit breakers when the G-05 is in operation; PC 29, "Monthly Gas Turbine and Auxiliary Diesel Load Test," OI 110, "Gas Turbine Operation," and NP 2.1.5, "Electrical Communications, Switchyard Access and Work Planning." Changes to the procedures were required to ensure that electrical fault currents did not exceed current interrupt ratings of the 13.8 kV system circuit breakers with G-05 connected to 13.8 kV bus H-01. Exceeding the current interrupt ratings of 13.8 kV system circuit breakers could result in damage to the 13.8 kV system.

Summary of 10 CFR 50.59 Evaluation: The potential for exceeding 13.8 kV circuit breaker interrupt ratings, given a unit trip in conjunction with a 13.8 kV system electrical

fault, was identified in Calculation 2005-0035, Rev. 0. The conclusions given in Calculation 2005-0035 state that disabling the automatic 4160 V fast bus transfer and limiting G-05 voltage to 101.0% of its rating will ensure that there are no overduties of circuit breakers at the 13.8 kV level in the event of a fault in conjunction with a unit trip. Disabling the 4160 V fast bus transfer during testing would cause one or both units to rely on natural recirculation in the event of a trip. Shutting down the G-05 by operator action if a unit trips or a 13.8 kV breaker trips, was selected as being equivalent to disabling the automatic bus transfer in Calculation 2005-0035. This allows controlled unit shutdowns rather than relying upon natural recirculation. The changes made did not result in abnormal operating requirements or abnormal operating conditions for G-05 and 13.8 kV Systems. Limits placed on operating parameters and the addition of assigned operator actions, if a fault or unit trip occurs, minimizes the risk of equipment damage without resulting in abnormal conditions. The procedure revisions did not introduce the possibility of a malfunction of an SSC important to safety with a different result. The procedure revisions did not introduce the possibility of an accident of a different type. Pursuant to paragraph (c)(1) of 10 CFR 50.59, this change did not require a license amendment. (EVAL 2005-007)

FSAR Change, Revision of FSAR 14.2.4, Steam Generator Tube Rupture

Activity Description: The radiological dose analysis for the Steam Generator Tube Rupture event described in FSAR 14.2.4 was revised to reflect a change to plant input values used in the analysis. The change to the radiological analysis for a SGTR event reflected a continuing release of 24-hour duration. This is an increase from the prior assumption that the releases from the event are terminated at 8 hours after event initiation. The change was made as a result of the determination that a natural circulation cooldown performed with a limiting case. A single steam generator may not be achievable within the previously assumed 8-hour duration. The revised analysis also corrected a previous non-conservatism in the modeled duration of an accident initiated iodine spike by increasing it from 1.6 hours to 4 hours.

Reaching RHR cut-in conditions when limited to a natural circulation cooldown, from a single steam generator, could require more than 8 hours, but less than 24 hours. This was demonstrated by Westinghouse calculation SE/FSE-C-WEP-45. Since the historical radiological analysis assumed an 8-hour cooldown, and conservatively maintained a high steaming rate for this shorter duration, this was considered acceptable. However, with the revised thermal/hydraulic analysis, a dose assessment for the longer duration cooldowns, that clearly shows bounding results, is desirable.

Summary of 10 CFR 50.59 Evaluation: Extending the duration of the accident-initiated iodine spike and the time when RHR cut-in conditions is assumed to occur are both conservative and representative of plant limitations. The resultant changes in dose consequences were calculated using the previously approved FSAR analysis methodology. The increases in site boundary, low-population zone and control room doses do not result in exceeding dose limit criteria. The increase in the site boundary and low-population zone doses is less than ten percent of the margin between the recalculated dose and the dose listed in FSAR 14.2.5. The control room doses are

within the GDC-19 limits. Pursuant to paragraph (c)(1) of 10 CFR 50.59, this change did not require a license amendment. (EVAL 2004-001)

ENCLOSURE 2

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Commitment Change Evaluations

Operability of Core Exit Thermocouples when RVH is Set: The original commitment stated "two of the core exit thermocouples should be left connected until just before the reactor vessel head lift and reconnected as soon as possible after the head was set in place." The revised commitment states "When the reactor coolant system (RCS) is not in reduced inventory, two independent core exit thermocouples may be removed from service as necessary to enable reactor vessel head (RVH) maintenance activities."

Justification For Change: Generic Letter 88-17, "Loss of Decay Heat Removal," was issued to licensees on October 17, 1988. The generic letter dealt with the loss of decay heat removal (DHR) when the RCS is in a reduced inventory condition. The generic letter contained eight recommended expeditious actions and six program enhancements for longer term actions. This commitment responded to Expeditious Action Item 3, RCS Temperature. The NRC recommendation was to "Provide at least two independent continuous temperature indications that are representative of the core exit conditions whenever the RCS is in a mid-loop (reduced inventory) condition..."

The licensee responded to the generic letter via submittals dated December 30, 1988, and August 25, 1991. While the generic letter and the WE letter dated December 30, 1988 both specifically address the requirements for mid-loop operation, the actual commitment did not specifically include the "mid-loop" phraseology. Therefore, if the commitment for operability of the core exit thermocouples (CETs) was literally complied with for all the RCS inventory conditions, the literal compliance and requirement conflicted with the ability to perform post-RVH set maintenance activities such as RVH dressing.

The revised commitment does not conflict with the recommendations of Generic Letter 88-17 or NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management." The revised commitment clarifies the commitment to maintain the CETs operational during reduced inventory conditions. (CCE 2005-001)

Vendor Technical Information Requirement: The original commitment was to Item 2.2, Part 2 of Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events." In WE letter dated March 27, 1987, the licensee committed to the vendor interface elements of the Vendor Technical Information Program (VTIP) to specifically note that the interface with the NSSS supplier included a formal exchange of technical information, including technical bulletins. The program included documentation of the receipt of the information back to the NSSS vendor. This commitment to provide receipt documentation can no longer be met.

Justification For Change: The NSSS supplier no longer transmits Vendor Technical Information (VTI) with a receipt form to be signed and returned. NMC maintains periodic contact with the NSSS supplier to ensure timely receipt of VTI. Specifically, paragraph 4.2.1 of administrative procedure NP 7.2.14, "Vendor Contact Program," states, "ECM shall maintain contact with the NSSS supplier to ensure VTI is received by PBNP in a timely manner. The NSSS supplier will be contacted on a monthly basis to verify that the VTI tracking database is current, or request a re-transmittal if required." (CCE 2005-002)

Chemical Interface Material Compatibility: The original commitment was in response to NRC Inspection Report IR 96-006 and the resultant NRC Enforcement Conference held on September 12, 1996. The licensee committed to revise the design control procedure to ensure that chemical interfaces with material are evaluated over the design life of the materials. This commitment is no longer required.

Justification For Change: PBNP has adopted NP 7.2.15, "NMC Fleet Modification Process," its associated procedure NP 7.2.19, "Design Inputs," and form QF-0515B "Design Input Checklist." The procedures and the form were based on the ASME NQA-1-1994, Subsection 3, design control requirements. The Design Input Checklist provides further clarification of this ASME NQA-1-1994, Subsection 3 requirement with respect to the "suitability of application" of materials.

Consideration of the basic material/chemical environment compatibility effects in a proposed design is an element of the design control process, which complies with the requirements of ASME NQA-1-1994. Therefore, maintaining a separate, specific commitment to require consideration of material/chemical environment compatibility effects is no longer necessary. (CCE 2005-003)