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Annual 10 CFR 50.59 Summary Report for 2006

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 evaluated in accordance with 10 CFR 50.59 during 2006. Enclosure 2 contains commitment change evaluations completed in 2006.

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

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Nuclear Management Company, LLC

Enclosures (2)

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC

ENCLOSURE 1

POINT BEACH NUCLEAR PLANT
ANNUAL 10 CFR 50.59 SUMMARY REPORT FOR 2006

Modifications
FSAR Changes and Other Evaluations

FSAR Change, Revision of FSAR Section 14.2, "Standby Safety Features Analysis" and Section 14.3, "Primary System Pipe Ruptures"

Activity Description: FSAR Sections 14.2 and 14.3 were revised to reflect the longer pump motor acceleration times for the 1P-15A&B and 2P-15A&B, safety injection pumps. Motor design data provided by the nuclear steam supply system (NSSS) supplier for the bolted fault project showed that the acceleration times for the safety injection pumps were longer than what was used in portions of the accident analysis. The letter documented that the impact of an additional 5.0 second delay (above the original design value of 5.0 seconds) in the high head safety injection pump spin-up to the Point Beach Nuclear Plant (PBNP) small and large break loss of coolant accident (LOCA) analysis, main steam line break (MSLB) core response analysis, and the LOCA and MSLB containment integrity analysis is acceptable.

Summary of 10 CFR 50.59 Evaluation: The addition of 5.0 seconds (above the original design value of 5.0 seconds) in the safety injection pump acceleration time has been evaluated for small and large break LOCA conditions, MSLB core response, LOCA and MSLB containment integrity analysis as documented by the NSSS vendor. The letter evaluated that the additional acceleration time will not have appreciable affect on the small and large break LOCA analysis. The letter states that the increase in safety injection pump acceleration time will have no effects on the results and conclusions of the analysis of MSLB core response. The letter also states that the increase in the acceleration time of the pumps will have no effect on the LOCA and MSLB containment integrity analysis. Therefore, the addition of 5.0 seconds to the acceleration time of the safety injection pumps will not result in a change in the results and conclusions of the PBNP accident analysis. (EVAL 2006-011)

FSAR Change, Revision of FSAR Section 14.2.5, "Rupture of a Steam Pipe"

Activity Description: Clarification was made to the discussion of assumptions used in the containment integrity analysis for a MSLB as described in FSAR Section 14.2.5. A discussion of the behavior of containment pressure and temperature after peaking and after auxiliary feedwater (AFW) flow is isolated to the faulted steam generator was added. Additionally, the effect of the failure to isolate AFW within the assumed time frame was clarified.

Summary of 10 CFR 50.59 Evaluation: Failure to isolate AFW flow would not affect the timing and magnitude of the peak containment pressure and temperature resulting from a MSLB. Failure to isolate AFW would not cause a second peak to occur later in the transient. After the peak pressure and temperature have occurred, heat removal from containment by the credited systems (spray and fan coolers) exceeds the heat input from steaming the faulted SG, even with continued AFW. As long as the faulted SG is not isolated from AFW, releases will continue from the steamline break and the containment pressure and temperature may remain elevated. Therefore, realignment of the AFW system is needed, but the conclusions from the MSLB containment response analyses are not affected by the timing of AFW isolation. (EVAL 2005-004)

FSAR Change, Revision of FSAR Section 9.11, "Sampling System"

Activity Description: The reference to ASME Section III, Class C, tube side, in Table 9.11-1 of FSAR Section 9.11 Revision 1, "Sampling System," was deleted. The code requirements for the sample heat exchanger were changed to ASME Section VIII for both the shell and the tube side.

Summary of 10 CFR 50.59 Evaluation: The original design of the sample system heat exchangers was to have the tube side designed to ASME Section III, Class C and the shell side to ASME Section VIII. This is per the Unit 1 data sheets that were supplied with the heat exchangers. The design specification for the heat exchangers states that the components shall be code stamped and assigned a national board number. The date on the specification is September 9, 1966. This specification applies to other auxiliary heat exchangers, not just the sample heat exchangers. On March 20, 1967, the requirements for the sample system heat exchangers were changed. Per the purchase order, the sample system heat exchangers fall outside the jurisdiction of ASME Section VIII and are not required to be code stamped or be assigned a national board number; only that the supplier provide a certificate stating that the associated code requirements had been met. The drawing for these heat exchangers was certified in May of 1967. Based on this, the latest version of ASME Section III that could have been used was 1965 with 1967 summer addenda. In this edition of the code, the requirements for a Class C vessel are that the vessel requirements of Section VIII apply to the vessel, with the stipulation that some special welds that may be used to form the vessel be made and inspected to the requirements of Section III. In addition, an ASME Section III Class C vessel was required to be N-stamped. However, the tube side of the heat exchangers do not contain any of these special welds and it had already been determined that the vessel would not be stamped. The only Section III-1965 (with 1967 addenda) requirements that apply to the tube side of the vessel are that the vessel shall be made to the requirements of Section VIII. Based on this code reconciliation it is acceptable to change the FSAR to reference Section VIII only. Changing the construction code of the tube side of the heat exchangers from ASME Section III Class C to ASME Section VIII modifies a method of evaluation. This method of evaluation is specified in the FSAR Section 9.11. However, based on the discussion above, the results of the new method are the same as the original. Therefore, changing the construction code of the tube side of the heat exchangers from ASME Section III Class C to ASME Section VIII does not result in a departure from a

method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses. (EVAL 2005-008)

Procedure Revision, Revision of Operating Instruction OI 35C, "480V Electrical Load Conservation"

Activity Description: A revision to Operating Instruction OI 35C, "480V Electrical Load Conservation," was performed. The purpose of this instruction is to provide direction for 480V electrical load conservation of buses that are susceptible to spurious tripping on overcurrent under the minimum voltage that could be expected to occur during limiting degraded voltage conditions. The specific OI 35C activity that is being evaluated is operational alignment of instrument air (IA) compressors.

Summary of 10 CFR 50.59 Evaluation: Aligning an IA compressor to "OFF," and relying on manual operator action to align it to "CONSTANT" to provide backup for the lead IA compressor does not result in more than a minimal increase in likelihood of failure of a system, structure or component (SSC) important to safety. A PRA evaluation was performed and demonstrated less than a minimal increase in the frequency of occurrence of an accident which has been previously evaluated. Aligning an IA compressor to "OFF" does not affect the failure probability of either the operating or non-operating "OFF" compressor. Equipment important to safety has IA accumulators installed with check valves to allow for continued equipment operation in the event of any IA system malfunctions. There are alarms and specific procedural steps enabling alignment of the "OFF" IA compressor to "CONSTANT" when the need arises. In addition, service air is available as a backup. Impact on plant systems has been considered and is minimal. The consequences of the failure of an SSC important to safety did not change in this case. There are no previously unanticipated failures created with this activity which would lead to the possibility of a new accident of a different type. Adequate defense in depth exists if an air compressor was to fail. (EVAL 2006-007)

Plant Modifications MR 04-018 and MR 04-019, Replacement of Purge Supply/Return Valves Inside Containment

Activity Description: Modifications, MR 04-018 and MR 04-019, replaced the purge supply and exhaust containment isolation valves 1(2)VNPSE-03213 (Purge Exhaust Fan Suction) and 1(2)VNPSE-03245 (W-2A/B Purge Supply Fan Discharge) with a blind flange during MODES 1,2,3, and 4. A spool piece will be installed when the containment purge system is needed. Three specific design functions are covered by this evaluation: 1) The ability of the containment purge valves to isolate during MODE 6; 2) the establishment of containment closure during reactor vessel head lift; and 3) the establishment of containment closure when the reactor coolant system (RCS) is <200°F.

Summary of 10 CFR 50.59 Evaluation: The containment purge penetration is required to be isolated for a fuel handling accident, loss of decay heat removal, and during reactor vessel head lift. Only one valve or one blind flange per penetration is a SSC

important to safety for each of the design functions addressed in this evaluation (RCS <200°F or MODE 6). The replacement blind flange does not interface with an SSC that can initiate an accident. The interfacing 120V electrical and IA systems have been abandoned or removed in accordance with applicable codes and regulations. The only SSC that this blind flange interfaces with that could initiate an accident that was previously evaluated is the polar crane (reactor vessel head drop). The blind flange is stored next to its penetration on a permanent trolley/rigging system. This trolley/rigging system is hung from the polar crane rail and is ultimately supported by containment. There is no significant effect on the crane rail and therefore frequency of the reactor vessel head drop accident is not affected. After this modification there will be one valve capable of being closed by an operable containment purge and exhaust isolation system for each penetration. This complies with the Technical Specifications and Technical Requirements Manual (TRM). Alternatively, the blind flange could be installed. In both allowed configurations (one blind flange or one valve per penetration), the likelihood of a malfunction is either the same as or less than the existing allowable configuration. No change will be made to TRM 3.9.4 with this evaluation. During a reactor vessel head lift the purge valves shall be required to be shut even if the blind flange is installed. The Unit 1 purge valves outside containment will be provided with a nitrogen backup system equivalent to that installed on Unit 2. The consequences of an accident are unchanged with this modification. Containment closure is maintained by one valve or a blind flange. One barrier is required for containment closure. Malfunction of the one required barrier for containment penetrations while the RCS is <200°F or MODE 6 has not been previously evaluated in the current licensing basis. The consequences of loss of instrument air will be the same after the modification. These modifications do not introduce the possibility of a new accident because the containment penetration does not interface with any system that could initiate an accident other than the polar crane where head drop has been evaluated. The penetration would need to close or be closed after an accident while the RCS is <200°. These modifications do not result in a change that would cause any system parameter to change. Therefore, the modifications do not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered.
(EVAL 2006-003-01)

Plant Modification MR 05-023, Removal of Safety Injection (SI) Test Line and Isolation Valves

Activity Description: The 2SI-839A, B, C, & D SI test line isolation valves were removed, along with portions of the associated test lines. The piping was removed near its attachment point on the SI accumulator discharge lines, through the valves, to the common header. Most unused piping supports and piping were removed from the plant. The pipe ends were capped, and drain valve 2SI-D19 was located to the cut location on the common header. All supporting components were removed including the instrument air supply valves, solenoid valves, position indicating switches and lights, control switches and most of the wiring. The instrument air piping was capped at the common header to the valves. All conduit and wiring to the common junction box inside containment was removed. All wiring between the control switches, terminal blocks, manually operated breakers and position indicating lights was removed. The wire

running from the C01 panel to the common junction box was abandoned in place. The 125 V DC manually operated breakers in the main control board have now become spares. The openings in control panel C01 Rear were patched with carbon steel plate and painted.

Summary of 10 CFR 50.59 Evaluation: This evaluation addresses the potential adverse affects to the capability to test the SI accumulator discharge check valves 2SI-867A&B, 2SI-842A&B, and stop valves 2SI-841A&B. The capability to use the SI-839 valves and their test lines to test these valves was removed. Statements in the FSAR documenting this will be removed. The 2SI-867A&B check valves are considered pressure isolation valves (PIV), and fall under the scope of Event V testing. The Event V test system is used to measure backleakage through these valves on a refueling outage frequency. The SI-839 valves currently are not used for this purpose. The current testing protocol meets all Event V and IST program requirements. The removal of the SI-839 valves and associated test lines will not affect the ability to test these PIV for backleakage. The 2SI-842A&B check valves are not considered PIV, and are not within the scope of the Event V order. However, 2SI-842A&B check valves are leak-tested quarterly using the SI pump and measuring a change in accumulator level. The SI-839 valves are currently not used to support this testing. The testing currently performed is adequate and appropriate for these valves, and meets all IST program requirements. The removal of the SI-839 valves and associated test lines will not affect the ability to test these check valves for back-leakage. The 2SI-867A&B and 2SI-842A&B check valves receive a forward flow exercise test during each refueling outage. The SI-839 valves and test lines cannot be used to perform this type of testing, and the current testing protocol meets all IST program requirements. The removal of the SI-839 valves and associated test lines will not affect the ability to test these check valves to pass forward flow. The 2SI-841A&B SI accumulator stop valves are stroke tested on a refueling outage frequency as required by the IST program. Stem-to-disc integrity is also checked by observing accumulator level change. These valves do not have a safety-related function to change position, nor do they have a safety-related function in the shut position. These valves are administratively locked open while at power. There are no procedural instructions that would use the SI-839 valves and test lines to verify that the SI-841 valves are open, and this type of testing is not required per the IST program. Removal of the SI-839 valves and associated test lines will not affect the ability to test these stop valves to pass forward flow. A revision to the FSAR is required.
(EVAL 2006-004)

Calculation Revision, Revision of Primary Auxiliary Building (PAB) Superstructure/Crane Calculation

Activity Description: The PAB crane, Z-015, was used to move loads up to and including 125 tons over the spent fuel pool. In order to allow this, the PAB superstructure needed to be evaluated for its ability to support the PAB crane loaded to its rated capacity of 125 tons, concurrent with a safe shutdown earthquake (SSE). Calculation PBNP-305336-Sol, Revision 1, was performed to demonstrate that the PAB superstructure is capable of supporting the PAB crane while carrying its capacity of 125 tons concurrent with an SSE. A 125 ton load bounds the loads associated with the TN-32PT (NUHOMS) system and the VSC-24 system.

Summary of 10 CFR 50.59 Evaluation: There are no methods of evaluation described, outlined or summarized in the PBNP FSAR for Class III structures. The methodology PBNP used for qualifying the PAB superstructure, as used in the calculation, was not a departure from a method of evaluation described in the PBNP licensing basis. This evaluation was required as a result of a potential "change to a method of evaluation", as was determined by 50.59 screening SCR 2006-0063. (EVAL 2006-006)

ENCLOSURE 2

POINT BEACH NUCLEAR PLANT
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Commitment Change Evaluations

Outboard Ventilation/Purge Valve CV-3212: The original commitment stated CV-3212 will respond to a CVI signal with either or both of the local limit switch bypass switches closed. With either of these switches closed, outboard valve CV-3212 will automatically open upon clearing of the CVI signal. However, its inboard ventilation/purge isolation valve and all other ventilation/purge valves will remain shut upon clearing of the CVI signal. The revised commitment states, outboard ventilation/purge valve CV-3212 will respond to a CVI signal. Outboard valve CV-3212 will not automatically open upon clearing of the CVI signal. The inboard ventilation/purge isolation flanges will remain shut upon clearing of the CVI signal.

Justification for Change: Outboard ventilation/purge valve CV-3212 will respond to a containment ventilation isolation (CVI) signal with either or both of the local limit switch bypass switches closed. With either of these switches closed, outboard valve CV-3212 will automatically open upon clearing of the CVI signal. These switches are no longer installed (removed by Modification IC-242). The outboard ventilation/purge valve CV-3212 will remain closed upon clearing of the CVI signal. This eliminates the concern that the valve could automatically go open when the containment isolation signal is reset.

This blind flange is no longer affected by a CVI signal. The blind flange will remain installed during MODES 1, 2, 3, and 4. The blind flange cannot automatically open when the containment isolation signal is reset. (CCE 2006-001)