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Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

NRC 2003-0046 10 CFR 50.59

June 30, 2003

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2
DOCKETS 50-266 AND 50-301
ANNUAL 10 CFR 50.59 SUMMARY REPORT FOR 2002

In accordance with the requirements of 10 CFR 50.59, Nuclear Management Company, LLC (NMC), licensee for Point Beach Nuclear Plant (PBNP), is submitting the 2002, 10 CFR 50.59 summary report.

The report contains descriptions of facility changes, tests and experiments, and commitment change evaluations that occurred during calendar year 2002.

Please contact us if you have any questions.

Sincerely,

A. J. Sayia
Site Vice-President

kml

Attachment

cc: Regional Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, NRR, USNRC
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ATTACHMENT 1

POINT BEACH NUCLEAR PLANT
ANNUAL 10 CFR 50.59
SUMMARY REPORT FOR 2002

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10 CFR 50.59 AND 10 CFR 72.48 CHANGES

PROCEDURE CHANGES

PBTP 111, 2SI-867B SI System to RC Loop B Cold Leg Check Flow Test (Cold) Using 2P-10A and 2P-10B RHR Pumps (Temporary)

This test will determine if the disc of check valve SI-867B is free to move full travel, therefore satisfying ASME section XI requirements for a full stroke test on a refueling frequency. RHR pump flow will be used to create an impulse that will drive the check valve disc against the valve body creating an acoustic signal that will be recorded and analyzed to verify disc full open travel. The test will be performed with the plant in mode 6, the reactor de-fueled, all affected unit fuel in the spent fuel pool and the refueling canal isolated. The refueling cavity will be flooded to at least 63 feet to ensure that adequate NPSH margin exists.

Summary of Safety Evaluation: Damage or degradation to the pumps is not expected. The RHR pumps will have been pre-tested to 1800 gpm prior to the test and verified to be in good condition. Test flow of 2000gpm will be only slightly higher. Previous testing has shown the pump capable of this flow without cavitation. The pump's flowrate and physical condition will be monitored by personnel during the test performance and vibration and hydraulic monitoring before and after will demonstrate that no pump degradation has occurred. Thus the likelihood of a pump malfunction is minimal. Pressure in the piping during the test will be well within design parameters. Some sections of ten inch piping will be exposed to flows higher than previously experienced. These sections normally see 3100 to 3200 gpm for portions of the cooldown procedure, during which no flow induced vibration problems been reported. Test flows will be around 4000 gpm for about 3 to 5 minutes. The short duration of the higher than normal flow condition indicates that failure due to flow-induced vibration is unlikely. Piping damage due to flow induced erosion and vibration is not expected to occur. The design basis limit for the RCS fission product barrier will not be exceeded.

Any possible accidents, malfunctions, failure modes and their consequences, namely rupture of the reactor coolant pressure boundary, Loss of AC power and pump degradation, have been previously addressed and are bounded by analyses in the FSAR, chapters 6.2, 9.2 and 14 respectively. The probability of an accident previously evaluated in the FSAR is not increased. At the time of the test the affected unit reactor core will be de-fueled with the plant in mode 6, all affected unit fuel in the spent fuel pool, the refueling canal isolated, and the RCS de-pressurized with the vessel head removed. Therefore there is no possibility of having an accident involving the reactor core and RCS pressure boundary. Fuel handling in the rector cavity will not occur during the test, thus the probability of this accident occurring will not be increased. The probability of having a Loss of AC power accident will not be increased by the test. Performance of PBTP 047 has demonstrated that excessive current will not be drawn by the pumps even at flows considerably higher than those to be used in PBTP 111. Starting current has been evaluated in NPM 2002-0171 to be within protective relaying setpoints. The probability of affecting the integrity of the reactor and core internals will not be increased since the affects of starting the two RHR pumps together will be much less than starting a reactor coolant pump. Initial conditions for the test include RCS temperature of

80 - 100 degrees Fahrenheit. This minimizes the heat exchange affect of the test on the component cooling system due to the close proximity of RCS and CC temperatures. In addition, no additional water sources have been created that might increase the consequences of any potential flooding to the primary auxiliary building. OP-3C and OP-7A administratively ensure that that refueling water does not get released to occupied areas of containment. As stated above, piping or valve damage is not expected to occur during the test. The proposed activity does not involve a method of evaluation described in the CLB. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (EVAL 2002-002)

EOP-0, Reactor Trip or Safety Injection (Unit 1 and Unit 2) (Permanent)
EOP-0.1, Reactor Trip Response (Unit 1 and Unit 2) (Permanent)
ECA-0.0, Loss of All AC Power (Unit 1 and Unit 2) (Permanent)
ARP C01 A I-9, Instrument Air Header Pressure Low, Unit 0 (Permanent)

The proposed activity is to make permanent procedure changes that were implemented in response to a condition that was identified where, with a procedure-directed operator action to control steam generator level (which could be accomplished by reducing flow through one or more AFW pumps), concurrent with a loss of instrument air (which would cause the AFW pumps' mini-recirculation valves to fail close), the potential existed for a simultaneous failure of the multi-stage high pressure AFW pumps due to very low or no flow through running AFW pumps.

Summary of Safety Evaluation: The auxiliary feedwater (AFW) system responds to plant transients and other initiating events or accidents. Failure of an AFW pump to start or run does not initiate a transient or an accident. The permanent procedure changes ensure adequate flow is maintained to the steam generators, thus providing sufficient cooling to the reactor coolant system to prevent overfill of the pressurizer and the possibility of a small-break LOCA due a relief valve failing due to water relief. Therefore, the frequency of occurrence of a small LOCA that may result from a stuck open pressurizer PORV is not increased. Operation of AFW pumps at reduced flows, and their ability to sustain repeated starts were considered. The minimum flows in the procedures meet vendor recommendations. Impact of periodic restarting of AFW pumps, motors and turbines was considered, along with electrical breaker and MOV operation. Operating procedures and operator training has addressed operator action to maintain minimum AFW flows or to secure the pump(s). Therefore, there is not more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the CLB. The FSAR accident analysis assumes a minimal AFW flow of 200 gpm, which is within the capability of one AFW pump at pressures greater than the peak calculated SG pressure in the analysis. The required minimum AFW flow capability will continue to be met by implementation of these procedure changes. Therefore there is no increase in radiological consequences due to accidents or malfunctions of SSCs important to safety. Even if operator error caused the failure of two pumps, then there is still one AFW pump available per unit to perform their design basis function. Therefore, the proposed activity does not create a possibility for an accident of a different type than any previously evaluated in the FSAR. The result of the malfunction of a pump, due to failing to restart or failing due to low flow, is that the AFW pump fails to provide feedwater to the steam generators which is the same result. Therefore, the proposed activity does not create the possibility of a malfunction with a different result. The FSAR analysis of the loss of normal feedwater and the loss of ac power events has demonstrated that the pressurizer volume remains below the acceptance limit (i.e., the

pressurizer does not overfill). The consequences of these events are analyzed in that no water relief occurs and no challenge to the pressurizer PORV or safety valves is created. Therefore no design basis limit for a fission product barrier is exceeded or altered. The procedure changes do not involve a method of evaluation; therefore the procedure changes do not result in a departure from a method of evaluation used in establishing the design bases or in the safety analyses. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (EVAL 2002-005)

PBTP 117, Test of Reactor Cavity Cooling System Requirements During Normal Plant Operation (Temporary)

PBTP 117 will obtain data to support isolating service water (SW) to the cavity coolers during normal plant operation. The test will show the temperature effects on nuclear instrumentation, control rod drive mechanism coils and concrete in the reactor cavity area utilizing a cavity cooling fan with SW isolated to its respective cooling coils.

Summary of Safety Evaluation: Higher temperatures are expected in the cavity area as a result of performing PBTP 117, but they will not exceed design limits. There are three temperature monitoring points that will be monitored to ensure that design temperature limits for control rod drive mechanisms, nuclear instrumentation or nozzle area concrete are not exceeded. The malfunction of the nuclear instrumentation or of a stationary coil is prevented due to the conservative temperature limits set for the data points. The RCCA drop accident is defined as a Catergory II Event and has been evaluated in the CLB. Current analysis concludes that the departure from nucleate boiling (DNB) resulting from the RCCA drop would still meet design due to DNB ratio being greater than the limit value. The CLB for containment pressure boundary is bounded by manual operator action to shut the manual isolation valves to isolate the system rupture. The thermal expansion issue (i.e. NRC GL 96-06) leading to thermal overpressurization of the isolated piping is not an issue, since the SW relief valves are still valved into the isolated piping boundary during the test. In addition, the coils will be partially drained prior to running the associated cavity cooling fan. Voiding in the isolated piping section is expected to accommodate any heating (thermal expansion) preventing relief valves lifting, thus eliminating any valve reseating concerns. The issue of a reactor trip supported by nuclear instrumentation: source range high level, intermediate range high level, power range high level (low setting) or power range high level (high setting) is accounted for by a temperature limit set lower than equipment design limits. The test has no impact on the CLB analysis, since expected elevated temperatures are maintained below design limits. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (EVAL 2002-006)

PBTP 118, 1SI-867B SI System to RC LOOP B Cold Leg Check Flow Test Using 1P-10A AND 1P-10B RHR Pumps (Temporary)

This test will determine if the disc of check valve SI-867B is free to move full travel, therefore satisfying ASME section XI requirements for a full stroke test on a refueling frequency. RHR pump flow will be used to create an impulse that will drive the check valve disc against the valve body creating an acoustic signal that will be recorded and analyzed to verify disc full open travel. The test will be performed with the plant in mode 6, the reactor de-fueled, all affected unit fuel in the spent fuel pool and the refueling canal isolated. The refueling cavity will be flooded to at least 63 feet to ensure that adequate NPSH margin exists.

Summary of Safety Evaluation: Damage or degradation to the pumps is not expected. The RHR pumps will have been pre-tested to 1800 gpm prior to the test and verified to be in good condition. Test flow of 2000gpm will be only slightly higher. Previous testing has shown the pump capable of this flow without cavitation. The pump's flowrate and physical condition will be monitored by personnel during the test performance and vibration and hydraulic monitoring before and after will demonstrate that no pump degradation has occurred. Thus the likelihood of a pump malfunction is minimal. Pressure in the piping during the test will be well within design parameters. Some sections of ten inch piping will be exposed to flows higher than previously experienced. These sections normally see 3100 to 3200 gpm for portions of the cooldown procedure, during which no flow induced vibration problems been reported. Test flows will be around 4000 gpm for about 3 to 5 minutes. The short duration of the higher than normal flow condition indicates that failure due to flow-induced vibration is unlikely. Piping damage due to flow induced erosion and vibration is not expected to occur. The design basis limit for the RCS fission product barrier will not be exceeded.

Any possible accidents, malfunctions, failure modes and their consequences, namely rupture of the reactor coolant pressure boundary, Loss of AC power and pump degradation, have been previously addressed and are bounded by analyses in the FSAR, chapters 6.2, 9.2 and 14 respectively. The probability of an accident previously evaluated in the FSAR is not increased. At the time of the test the affected unit reactor core will be de-fueled with the plant in mode 6, all affected unit fuel in the spent fuel pool, the refueling canal isolated, and the RCS de-pressurized with the vessel head removed. Therefore there is no possibility of having an accident involving the reactor core and RCS pressure boundary. Fuel handling in the rector cavity will not occur during the test, thus the probability of this accident occurring will not be increased. The probability of having a Loss of AC power accident will not be increased by the test. Performance of PBTP 047 has demonstrated that excessive current will not be drawn by the pumps even at flows considerably higher than those to be used in PBTP 118. Starting current has been evaluated in NPM 2002-0171 for unit two and NPM 2002-0363 for unit one to be within protective relaying set-points. The probability of affecting the integrity of the reactor and core internals will not be increased since the affects of starting the two RHR pumps together will be much less than starting a reactor coolant pump. Initial conditions for the test include RCS temperature of 80- 100 degrees Fahrenheit. This minimizes the heat exchange affect of the test on the component cooling system due to the close proximity of RCS and CC temperatures. In addition, no additional water sources have been created that might increase the consequences of any potential flooding to the primary auxiliary building. OP-3C and OP-7A administratively ensure that that refueling water does not get released to occupied areas of containment. As stated above, piping or valve damage is not expected to occur during the test. The proposed activity does not involve a method of evaluation described in the CLB. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (EVAL 2002-007)

PBTP 119, Control Room Ventilation Test - Pressure Boundary (Temporary)

PBTP 119 will test the control room emergency ventilation system to gather data for future system optimization. This test will place the control room emergency filter system (CREFS) in a modified emergency mode consisting of a mixture of filtered outside air and filtered return air from the control room. The CREFS will be considered inoperable

during the performance of this test and the requirements of Technical Specification 3.7.9 will be observed. Data will be obtained during the performance of the test.

Summary of Safety Evaluation: PBTP 119 will be performed well within the Allowed Outage Time of Technical Specification 3.7.9. Following the test it will be restored to an operable condition as required by the TS. This test has been evaluated for any potential affects on the CLB and existing assumptions and has no adverse consequences on the CLB, safety-related or important to safety SSCs or for control room operations personnel. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (EVAL 2002-008)

MODIFICATIONS COMPLETED IN 2002

MR 98-024*O, Add Service Water Copper Ion Injection System

This modification adds a new copper ion injection system to the service water system to inhibit the growth and attachment of zebra mussels. The added injection system is non-safety related and isolated from the service water system by seismically qualified check valves that will be tested under the IST program.

Summary of the Safety Evaluation: The modification will not increase the probability of occurrence of an accident or event previously evaluated in the CLB. Service water is not an initiator of any of the accidents analyzed in the FSAR. Failure of the non-safety piping of the added injection system will result in a leak of 75gpm into the service water pump cubicle, which is well within the existing floor drain capacity. The modification does not impact the Fire Protection Evaluation Report. The modification will not increase the probability of a malfunction of equipment important to safety previously analyzed in the FSAR. The new injection system has no safety-related functions. The added flow to the service water system is less than 1% of the service water system flow and therefore will have an insignificant impact on service water temperature and pressures. The injected ion, Cu^{+2}, is known to accelerate corrosion rates of Cu-Ni alloys but, at the intended concentrations of 5-10 ppb above natural background, there will not be a measurable increase in corrosion. The modification will not increase the radiological consequences of an accident, event or malfunction of equipment important to safety previously analyzed in the CLB. The service water system is not a source of radioactive materials and is not an initiator of analyzed accidents or events with radiological consequences. The service water system is relied upon for proper operation of equipment used to mitigate the consequences of accidents or events. This modification does not change service flows or pressures to safety-related equipment. This modification does not create any new unmonitored release paths and does not involve high energy lines.

This modification does not create the possibility of an accident or event of a different type than any previously evaluated in the CLB. The service water system does not initiate events except for flooding, which is described above. Potential failures are bounded by failures previously considered in the CLB. The modification will not create the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated. Any postulated failure of new equipment or equipment used during installation is bounded by failures previously evaluated in the CLB. The proposed modification does not affect the margin of safety defined in the Technical Specification bases. The new piping and components do not affect the operability of the service water system. The safety classification and seismic class of new components is consistent with current system design. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (SE 2001-0020)

MR 99-040, Refueling Water Storage Tanks (RWST) Seismic Upgrade

The scope of this modification is to add new anchor details to the base of each RWST and reinforce most of the existing anchor seats.

Summary of Safety Evaluation: This modification adds new anchor details and reinforcing to the existing anchor seats. This increases the capability of the RWST base to resist an earthquake.

The modification will not affect the tank's storage capacity, chemical composition of the water, safety function, or the method by which the safety function is performed. Therefore, the final configuration will not increase the probability of an accident or event occurring, the possibility of a malfunction of equipment important to safety, create an accident or malfunction of equipment important to safety different from those analyzed in the CLB, affect the consequences of an accident or reduce the margin of safety defined in the Technical Specification bases. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (SE 2000-0044)

MR 00-035 and MR 00-036, Reroute Vacuum Priming Pump Drains to Atmospheric Blowoff Tanks

The scope of these modifications are to reroute the vacuum priming pumps (P-4I,IP-41, and 2P-41) drainage to Outfalls 001 and 002 as directed by the Wisconsin Pollution Discharge Elimination System (WPDES) permit. The drain previous to the modification went to the roof drain system, which empties on the beach. This drain path is plugged during chlorination because there is no dechlorination addition to this flow path. The overflow from the temporarily plugged lines will drain to the turbine hall sump system. FSAR figures will be updated to reflect the change.

Summary of Safety Evaluation: The drainage is collected after leaving the vacuum priming pumps, and so will not have an effect on pump operation. The drains will enter the circulating water loop seal downstream of the atmospheric blowoff tanks, and are at atmospheric pressure. None of the equipment involved in the reroute involves SSCs evaluated in the CLB accidents or events. Tapping into the atmospheric blowoff tanks occurs during unit outages, and the circulating water loop seal is below the floor of the turbine hall, so there is no possibility of flooding during installation. The reroute does not affect the operation of plant equipment. The water that is rerouted is from a point in the service water system that is not downstream of any service water components, so there are no radiological effects associated with the reroute or any equipment involved in the reroute. None of the equipment involved is important to safety. The routing is downstream of any component important to safety, and therefore cannot create the possibility of an accident or event of a different type than any previously evaluated in the CLB. There are no effects upon Technical Specifications. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (SE 2001-0058)

MR 00-076, South Service Building and Turbine Building Front Office Chiller Replacement

The scope of this modification is to replace the south service building chiller (HX-028A) and the turbine building front office chiller (HX-028B). The new chillers will have added cooling capacity and enhancements for maintainability. New electrical breakers will be installed for both new chillers. The south service building chiller will be supplied with a new power cable. The new chillers will be placed on the existing foundations. The control loops for each chiller will remain the same. The service water supply and return configurations for the existing chillers will be modified to accommodate the new chillers. This will require a change to the FSAR figure depicting the service water system.

Summary of Safety Evaluation: The chillers are non-QA, non-safety related, seismic class III. They will require half the service water flow as the existing chillers and service water modeling of this reduced flow shows no adverse effect. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (SE 2001-0053)

MR 01-082, Unit 2 Steam Generator HX-1A Main Feedwater 6 Inch Bypass Line Relocation

The scope of this modification is to relocate the tap for the unit 2, steam generator A (2HX-1A) 6 inch main feedwater bypass line (6"-DB-1) from its present location upstream of the unit 2 leading edge flow meter (LEFM 2FE-3 110) to immediately downstream of the LEFM. This obviates the need for isolation of the bypass line precision feedwater flow measurements and will permit using the LEFM as a continuous online precision feedwater flow measurement.

Summary of Safety Evaluation: Relocation of the 6 inch main feedwater bypass line tap will be performed in accordance with the original code of construction and design guidelines for this piping class, therefore the potential for failures following the modification remain unchanged from the original installation. No new valves are being added by the modification and the basic configuration of the existing line is being maintained, therefore, no new failure modes or impact on the function or operation of the 6 inch main feedwater bypass line or main feedwater bypass valve 2CS-480 to unit 2 A steam generator are being introduced. Except for fire protection, no equipment important to safety is in the area where the modification is being performed. The modification has no impact on the operability of the fire protection equipment during and following installation.

The modification does not add new components nor introduce any new failure modes to the normal feedwater system. No increase in the probability of an accident or event previously evaluated in the CLB (loss of normal feedwater) nor an accident or event of a type different than previously evaluated in the CLB been identified with implementation of the modification. In addition, no increase in the probability of a malfunction of equipment important to safety as evaluated in the CLB or a malfunction of a different type than previously evaluated has been identified. Since the FSAR identifies no radiological consequences with loss of normal feedwater accidents and there has been no new components or failure modes introduced with the implementation of this modification which could increase in the probability of an accident, event, or malfunction of equipment important to safety, there is no increase in the radiological consequences associated with these events. There are no Technical Specifications associated with the main feedwater piping and no new failure modes are being added by the modification that could change the margin of safety as defined in the basis for any Technical Specification. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (SE 2001-0055-01)

MR 01-085 and MR 01-086, Debris Filter on Condensate Cooler Circulating Water Piping

The scope of these modifications are to add cone type automatic debris filters up stream of each condensate cooler to reduce frequent macro fouling of the condensate cooler

and the resulting inability to control generator hot gas temperatures. Each new filter will require electric power, instrument air (IA) and a discharge line to the circulating water return down stream of the condensate cooler. A local control panel per filter will provide fully automatic monitoring and operation of the debris filtration system. The modification affects FSAR figures 10.6-1 and 10.6-1A

Summary of Safety Evaluation: This modification will not affect the ability of the circulating water system to provide flow to the condenser or affect the ability of the condensate cooler to cool condensate. Condensate flow paths remain unchanged by this modification. It will also not affect the availability of the main condenser before, during or after an accident. Failure of the new filter (plugging) has the same effect as the existing configuration. The filter has a by-pass installed so in the event of malfunction causing high differential pressure across the filter screen an automatic by-passes opens ensuring flow to the HX and protecting the filter. The new filter has been designed to collect debris and pass it to the down stream side of the condensate cooler. The down stream piping discharges to the lake therefore no equipment downstream will be affected by this modification. This new load will be powered from a non-safety related power supply which would not be required during a design basis accident. IA will also be supplied to the skid. IA is non-safety related, non-QA and non-seismic and is not required during a design basis accident.

This modification does not increase the probability of occurrence of an accident or event or increase the probability of a malfunction of equipment important to safety previously evaluated in the CLB. The modification does not increase the radiological consequences of an accident, event, or malfunction of equipment previously evaluated in the CLB. The modification does not create the possibility of an accident or malfunction of a type not previously considered. The modification does not affect any steam and power conversion components or systems as described in TS 15.3.4. Therefore this modification does not activity reduce the margin of safety defined in the basis for any Technical Specification. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (SE 2001-0050-01)

MR 01- 101, Optimize New 900 MHz Radio System

The scope of this modification is to optimize the new Motorola 900 MHz radio system by installing a high gain antenna on the roof of the unit 2 facade, a fifth dispatch console position in the operations support center (OSC), and by eliminating known radio frequency (RF) dead zones around the plant. The high gain antenna will allow radio system coverage to be extended the necessary ten miles from the plant so that offsite radiation protection personnel can communicate with onsite personnel during an emergency or drill. The fifth dispatch console position in the OSC will enhance Emergency Plan implementation. New antennas will be installed to eliminate known RF dead zones around the plant. Old radio system E (OPS 3), which is currently providing the ten-mile offsite radio coverage, will remain in service until the high gain antenna is ready to be connected. All other radio equipment is new and will have no impact on the in-service radio system. Post maintenance testing will ensure that proper coverage has been achieved. The in-plant coverage tests will not be performed in areas that are known to contain RFI sensitive equipment. The scope of this work is non-safety related.

Summary of Safety Evaluation: New antennas are being installed in new locations, but the areas have been evaluated for RFI sensitive equipment with no RFI sensitive

equipment found. The radiated power from the new antennas will be less than the most conservative equipment susceptibility limits established in EPRI technical report TR-102323. In addition, portable radios had been used in these locations with the old VHF radio system with no adverse effects. Some of the new conduits required will be routed through seismically qualified areas. QA hardware will be used to ensure no interaction with safety related equipment and to meet seismic two-over-one criteria. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (EVAL 2001-001-01)

FSAR CHANGES AND OTHER EVALUATIONS

FSAR Section 7.5, Operating Control Stations

Before the upgrade of the PPCS under the various parts of MR 98-002, the PBNP licensing basis was that the safety parameter display system (SPDS) was seismically qualified. This safety evaluation evaluates the downgrade of the new PPCS 2000 computer and associated signal multiplexers to a "non-seismic" design requirement. The old PPCS met the requirements of Reg. Guide 1.97 1 Type A variable for core exit thermocouples. This variable has been moved to new dedicated seismically qualified and safety related recorders in the ASIP panels (IC20 and 2C20). As a result, the PPCS and the input signal multiplexers can now be downgraded to a non-seismic design. This change was reflected in the June 2002 update to the FSAR (FCR 02-004 – supported by this safety evaluation).

Summary of Safety Evaluation: The PPCS 2000 provides information to the operators in a convenient format. The PPCS 2000 is not required for safe operation of the plant since all the critical information needed by operators is available on seismically qualified indications. The seismic downgrade of PPCS 2000 does not change the set of instruments that provide critical safety system parameter information available to the control room operator. Seismically qualified instruments are available to the operators which provide all the critical parameters if the PPCS 2000 is not available. Operators are trained to use the seismically qualified instruments when SPDS is not functional. Since the PPCS 2000 continues to meet the regulatory requirements and seismically qualifications indications are available upon loss of PPCS 2000 this change does not result in more than minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety. The PPCS 2000 displays information only; PPCS cannot increase the frequency of an accident or create a possibility for an accident of a different type. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (EVAL 2002-001)

FSAR Section 11.2, Gaseous Waste Management System (WG)

The proposed change is to increase the allowable letdown gas stripper flow rate from 80 gpm to 90 gpm. The change is being made to accommodate an increase in letdown flow through the 40 and 80 gpm orifices due to a return to the normal operating pressure of 2250 psig. The gas strippers are specifically used to facilitate degassing of the RCS during shutdown operation with either two 40 gpm orifices aligned or one 80 gpm orifice. This change has been processed under FCR 02-020.

Summary of Safety Evaluation: Increasing the flow rate though the gas stripper is not an initiator of any new malfunctions and no new failure modes are introduced. The support systems to the gas stripper have sufficient margin available to accommodate increased letdown flow. The increased flow does not affect the ability of the support systems to perform their intended safety functions. Specifically during a LOCA the letdown line is isolated removing the heat load from component cooling and the radwaste steam isolation function is not hindered by using the available radwaste steam margin. The increased flow rate through the cryogenic gas decay tanks has a less than minimal increase on the dose consequences for the cryogenic gas decay tank rupture. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (EVAL 2002-003)

NOTE: EVAL 2002-003 also established the basis for revising FSAR Section 14.2.3, Accident Release – Waste Gas using the uprated letdown stripping flow of 90 gpm. The revisions to Section 14.2.3 have been superceded by further analysis for the power uprate. These analyses assumed a letdown stripping flow that bounds the proposed revisions in this Safety Evaluation. Therefore, EVAL 2002-003 establishes the basis for revising the description of the Waste Gas system in FSAR Section 11.2.

FSAR Section 14.1.9, Loss of External Electrical Load
FSAR Section 14.1.10, Loss of Normal Feedwater
FSAR Section 14.1.11, Loss of All AC to the Station Auxiliaries

The purpose of this safety evaluation is to support revisions to FSAR 14.1.9, 14.1.10, and 14.1.11 Analysis for 50# Steam Generator to MSSV Pressure Drop. These changes have been processed under FCR 02-022.

Summary of Safety Evaluation: A non-conservative input was used in the accident analysis for the Loss of External Electrical Load (FSAR 14.1.9), Loss of Normal Feedwater (FSAR 14.1.10), and Loss of All AC to the Station Auxiliaries (FSAR 14.1.11) transients. The non-conservative input is the pressure drop from the steam generator to the main steam safety valves (MSSV) during full flow conditions. A calculation determined that the pressure drop under full flow conditions is 44 psig. The input in the current analysis is 20 psig, and the revised input is conservatively assumed to be 50 psig, bounding the calculated value of 44 psig. Westinghouse has performed a re-analysis/evaluation using the new pressure drop, and a revised steam generator low-low water level trip setpoint, and has generated new results for these transients. All acceptance criteria and design basis limits for fission product barriers continue to be met. The FSAR has been updated with the results of the Westinghouse analysis. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (EVAL 2002-004)

FSAR Section 14.2.3, Accident Release – Waste Gas

This EVAL supports changes to FSAR section 14.2.3, waste gas system accident analysis dose consequences due to a revised primary coolant source term which implements the impact an uprated core operating level of 1683 MWt and conservative changes to various input parameters. A change in power is not being proposed in the FSAR change, and the change only serves to bound any future changes to the rated thermal power level, including a measurement uncertainty recapture uprate (1.4% of 15 18.5 MWt). Accident analyses for 3 of the 5 scenarios described in this section are revised. The tank rupture scenarios subject to the implementation of an uprated thermal power source term are gas decay tank (GDT), volume control tank (VCT), and charcoal-filled delay/decay tank (CDT). The FSAR change has been processed under FCR 03-012.

Summary of Safety Evaluation: Implementation of a revised release source term for the accident scenarios described in FSAR 14.2.3 does not affect the frequency of the accident or likelihood of occurrence of a malfunction of an SSC important to safety because the proposed changes to the source term are not initiators to the accidents and no new failure modes are introduced. The revised source term does increase the dose consequence for each of the tank rupture scenarios; however, the increases are less

than a minimal increase in dose consequences. In summary, the GDT whole body dose was increased from 0.77 rem to 0.82 rem; the VCT whole body dose was increased from 0.025 rem to 0.32 rem and the thyroid dose from 0.022 rem to 0.13 rem; and finally, the CDT whole body dose as increased from 0.1 rem to 0.18 rem. The implementation of the uprated thermal power source term does not create the possibility for an accident or malfunction of a different type because the changes to the input parameters do not introduce a new failure mode or mechanism. The implementation of the uprated thermal power source term for the waste gas system accident analysis does not result in exceeding or altering a design basis limit for a fission product barrier. Finally, the uprated thermal power source term and dose calculations do not result in a departure from a method of evaluation as described in the FSAR because development of the source term and dose calculations were done consistently with the information as provided in the FSAR. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (EVAL 2002-011)

Changes to Bases Document for TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System

The existing Bases for Technical Specification 3.4.12 contains inconsistent statements regarding the status of accumulator discharge MOV supply breakers when the accumulators are required to be isolated. For example; APPLICABLE SAFETY ANALYSES states the breakers must be "fixed in their open positions," then later states that the LTOP requirements are met by " Deactivating the accumulator discharge isolation valves in their closed positions." The LCO discussion states that the valves must be "closed and immobilized." The SURVEILLANCE REQUIREMENTS discussion states that the valves are verified "closed and locked out." The proposed change replaces the various equipment status descriptions with a single statement in the surveillance section that will state that an accumulator is considered isolated when the discharge isolation valve is closed and power has been removed from the valve operator under administrative controls.

Summary of Safety Evaluation: The activity (change to status controls and surveillance associated with an isolated accumulator) does not affect FSAR Chapter 14 evaluated accidents. The activity does have a potential effect on the frequency of occurrence of a mass input type of transient (i.e., accumulator discharge) causing an overpressurization of the RCS at temperatures less than the LTOP enabling temperature. The activity would impose less stringent controls on the discharge isolation valves than those possibly construed to presently exist (i.e., equipment status controlled with locks). The proposed requirements for the accumulator discharge isolation valves will be comparable to the LTOP requirements that exist for the SI pumps and will ensure that no single malfunction or action will result in the accumulator discharging into the RCS. Locks are judged to provide minimal additional protection against inadvertent or improper equipment operation over that provided by other administrative controls and the 12 hour verification of equipment status required by SR 3.4.12. Therefore, it is concluded that the proposed activity will not result in more than a minimal increase in the frequency of occurrence of an accident (or event) previously evaluated in the CLB. The activity does not modify any plant equipment, nor cause any equipment to be operated outside of its design ratings. The activity does not introduce the possibility of a change in the likelihood of a malfunction and does not affect equipment used to mitigate the consequences of an overpressurization event. The activity does not change the consequences of a malfunction of SSC, nor does it create a new initiating event, introduce a new failure

mechanism, or create a new condition, which would exceed design limits of any equipment important to safety. The activity does not replace an automatic action with manual action, and does not introduce new operator actions, which may be prone to error. The activity does not alter any of the P/T limits established in the PTLR, nor any of other established limits for a fission product barrier. TS 3.4.12 maintains controls to preclude overpressurization of the RCS caused by spurious actuation of the accumulator discharge isolation valves. Therefore, the activity will not result in more than a minimal increase in the likelihood of occurrence or consequences of an previously evaluated accident or malfunction of SSC, does not create the possibility of an accident or malfunction of SSC of a different type than previously evaluated, does not result in a design basis limit for a fission product barrier to be exceeded or altered, and does not result in a departure from a method of evaluation described in the CLB. Pursuant to paragraph (c) (1) of 10 CFR 50.59, this change does not require a license amendment. (EVAL 2002-010)

COMMITMENT CHANGE EVALUATIONS

Control Room Radiological Dose Analyses An April 6, 2001, letter to the NRC stated PBNP will revise and submit radiological dose analyses for the control room, and a license amendment proposal as necessary to demonstrate continued conformance to the regulatory requirements and the Point Beach Nuclear Plant licensing basis needs a due date extension from February 1, 2002, to February 28, 2002.

Justification for Change: Additional evaluations, requested by NMC to improve the quality of the radiological dose analyses, resulted in a later completion of the final dose analyses by our vendor, Westinghouse, than initially planned. This has slightly delayed the review of the completed analyses by onsite staff and necessitated extension of the submittal date to the NRC. (CCE 2002-001)

Disposal of Contaminated Sewage Sludge, A letter from Wisconsin Electric to the NRC dated October 8, 1987, stated that prior to disposal, the waste stream will be monitored to determine the physical and chemical properties of the sludge. The change allows characterization of the sludge on an annual basis.

Justification for Change: Subsequent to the October 8, 1987, submittal the Wisconsin Department of Natural Resources (WDNR) issued a new Wisconsin Pollutant Discharge Elimination System (WPDES) permit to Point Beach Nuclear Plant (PBNP) on November 30, 1988. Both the new WPDES permit and the PBNP Sludge Management Plan specify an annual required frequency for the evaluation of the sludge characteristics. (CCE 2002-002)

Disposal of Contaminated Sewage Sludge, The above referenced letter also stated that radionuclide concentrations in the sludge shall be determined prior to each disposal by obtaining three representative samples from each of the sludge storage tanks. The change states that the sewage sludge may be disposed of by any legal method without gamma isotopic analysis (GIA).

Justification for Change: Pursuant to NRC guidance, the sludge is clean if no licensed materials are found when analyzed under conditions necessary to achieve the environmental Low Level of Detection (LLDs) (NRC HP Position Papers 221). Clean sludge is not under NRC jurisdiction and may be disposed of by any legal method without prior radioanalysis . Therefore, if the sludge is clean and there is no pathway to the Sewage Treatment Plant (STP) from the Radiological Controlled Area (RCA), or pathways are administratively controlled to prevent the transfer of licensed materials to the STP, there is no need to analyze the sludge to any disposal. Engineering modifications and administrative controls have eliminated the pathways from the RCA to the STP. (CCE 2002-003)

Annual Volume of Sludge Disposal, the above referenced letter also stated the annual disposal rate for each of the approved land spread sites will be limited to 4,000 gallons/acre, provided WDNR chemical composition, NRC dose guidelines, and concentration and activity limits are maintained within the appropriate values. The change states the limitation on the annual volume of sludge disposal per acre contained

is modified to allow unlimited disposal provided NRC and WDNR requirements and modifications are met.

Justification for Change: The original requirement to limit sewage sludge disposal to 4,000 gallons per acre was based on the assumption that the sewage sludge is contaminated with Co-58 at a concentration that it is 10% of the 10 CFR 20 Appendix B, Table 2, Column 2 value. Past sewage sludge disposal experience has shown the sludge may or may not be contaminated and, if it is, at concentrations far below 10% of the regulatory value when performed prior to each sewage sludge disposal. With the removal of some of the land spread sites due to their use as a storage site for dry storage of spent fuel, this requirement is limiting the ability to dispose of the sewage sludge on the remaining approved land spread sites. This change was evaluated under Safety Evaluation Report (SER) 95-057, "Removal of Licensee Commitment Involved with Sewage Sludge Disposal," dated April 20, 1995. (CCE 2002-004)

G01/G02 Relay Outliers: Provided an update to GL 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issues (USI) A-46," via letter dated December 21, 1999. The letter stated that, "USI A-46 seismic outliers for equipment at Point Beach, as discussed in GL 87-02, are planned to be resolved by the year 2000, except for outliers associated with the refueling water storage tanks (RWSTs), one containment isolation valve, and the (emergency diesel generator) G01/G02 relays. "The G01/G02 essential relay outliers are expected to be resolved by 2002." The commitment change revises this schedular commitment to March 28, 2002.

NOTE: Installation Work Plan (IWP) 98-114*Test, associated with Work Order WO 9935711 and Modification MR 98-114, was completed in March 2003. The IWP documents the testing of the new installed relays is complete and satisfactory and that all items that need to be completed prior to acceptance have been completed.

Justification for Change: The non-qualified seismic seismic qualification users group (SQUG) relay replacement for emergency diesel generator (EDG) G-02 was completed in 2002. Due to emergent issues associated with the G-02 relays, it is necessary to postpone completion for EDG G-01 until 2003.

Operability Determination CAP002694 provides a justification for continued operation of the plant with the outlier relays. The reason the relays are outliers is not related to aging of the relays. Based on this justification, it is acceptable to continue to operate the plant until March 28, 2003, with the current outlier relays. (CCE 2002-006)