PREFACE

This Annual Results & Data Report for 1984 is submitted in accordance with Point Beach Nuclear Plant, Unit Nos. 1 & 2, Technical Specification 15.6.9.1.B and filed under Docket Nos. 50-266 & 50-301 for Facility Operating License Nos. DPR-24 & DRP-27, respectively.

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1.0 INTRODUCTION

The Point Beach Nuclear Plant, Units 1 and 2, utilize identical pressurized water reactors rated at 1518 MWt each. Each turbine-generator is capable of producing 497 MWe net (524 MWe gross) of electrical power. The plant is located 10 miles north of Two Rivers, Wisconsin, on the west shore of Lake Michigan.

2.0 HIGHLIGHTS

2.1 Unit 1

Highlights for the period January 1, 1984, through December 31, 1984, included the completion of the steam generator replacement project. This project was in its ninety-first day at the beginning of the period, and continued for 99 days into 1984, returning to service on April 9, 1984. There was one brief outage to clean 19 incore flux thimbles and to replace reactor coolant loose parts monitoring system cable in containment. Unit 1 operated at an average capacity factor of 72.9% and a net electric/thermal efficiency of 34.5%. The unit and reactor availability were 72.7% and 73.1%, respectively. Unit 1 generated its 42 billionth kilowatt hours (gross) on May 30, 1984; its 43 billionth kilowatt hour on August 21, 1984; and its 44 billionth kilowatt hour on November 10, 1984.

2.2 Unit 2

Highlights for the period January 1, 1984, through December 31, 1984, included a 54-day refueling outage, a brief outage to clean flux thimbles and repair moisture separator reheater tubes and a brief outage to perform preventative maintenance to primary system hydraulic support. Unit 2 operated at an average capacity factor of 82.4% and had a net electric/thermal efficiency of 33.9%. The unit and reactor availability were 84.5% and 86.0%, respectively. Unit 2 generated its 41 billionth kilowatt hour (gross) on January 4, 1984; its 42 billionth kilowatt hour on March 26, 1984; its 43 billionth kilowatt hour on June 21, 1984; and its 44 billionth kilowatt hour on September 11, 1984.

3.0 FACILITY CHANGES, TESTS & EXPERIMENTS

3.1 Amendments to Facility Operating Licenses

During the year 1983, there were 13 license amendments issued by the U. S. Nuclear Regulatory Commission to Facility Operating License DPR-24 for Point Beach, Unit 1, and 12 for Point Beach, Unit 2. These license amendments and changes are listed by date of issuance and are summarized below:

3.1.1 03-11-83, Amendment 69 to DPR-24, Amendment 74 to DPR-27

These amendments deleted the Appendix "B" Environmental Technical Specification which pertain to the nonradiological water quality related requirements as required by the Federal Water Pollution Control Act Amendments of 1972.

3.1.2 04-15-83, Amendment 70 to DPR-24, Amendment 75 to DPR-27

These amendments addressed Technical Specification testing requirements for the containment airlock doors to achieve compliance with Appendix "J" to 10 CFR 50.

3.1.3 04-04-83, Amendment 71 to DPR-24, Amendment 76 DPR-27

These amendments consisted of changes to the Technical Specifications to allow repair of degraded steam generator tubes by sleeving, established primary coolant limits for iodine concentrations and surveillance frequency, and established a plugging limit for sleeved tubes.

3.1.4 04-15-83, Amendment 72 to DPR-24, Amendment 77 to DPR-27

These amendments modified the Specification regarding the frequency for conducting independent audits of the emergency preparedness program from once every 24 months to annually in accordance with the requirements of 10 CFR 50.54(f).

3.1.5 05-04-83, Amendment 73 to DPR-24, Amendment 78 to DPR-27

These amendments modified the Technical Specifications to allow temporary isolation of the shared motor-driven auxiliary feedwater pumps for a unit during periods of startup, shutdown and surveillance testing of the other unit provided that the turbine-driven auxiliary feedwater pumps are operable and capable of automatically delivering flow to the steam generators.

3.1.6 <u>09-30-83, Amendment 74 to DPR-24, Amendment 79 to DPR-27</u>

These amendments revised the degraded grid voltage relay setpoint and associated time delay as presented in Table 15.3.5-1 of the Technical Specifications.

3.1.7 <u>09-30-83, Amendment 75 to DPR-24</u>

This amendment approved the steam generator repair program for Point Beach, Unit 1, and required as a condition of license that the repair operations be conducted in accordance with those commitments identified in the approved repair report and reflected in the Staff's safety evaluation.

3.1.8 <u>10-06-83, Amendment 76 to DPR-24, Amendment 80 to DPR-27</u>

These amendments incorporated various administrative changes in the Technical Specifications in order to clarify terminology used in a limiting condition for operation, clarify language relating to a periodic calibration interval and correct specific portions of the specifications and bases.

3.1.9 <u>10-05-83, Amendment 77 to DPR-24, Amendment 81 to DPR-27</u>

These amendments revised the Technical Specifications to permit locating the spent fuel pool neutron absorber surveillance specimens adjacent to the spent fuel divider wall.

3.1.10 <u>10-31-83, Amendment 78 to DPR-24, Amendment 82 to DPR-27</u>

These amendments revised the loss of voltage relay setpoints and associated time delays in Table 15.3.5-1 of the Technical Specifications.

3.1.11 <u>12-12-83, Amendment 83 to DPR-27</u>

This amendment allowed a one-time relaxation of the length of time that the backup component cooling water heat exchanger may be out of service for maintenance.

3.1.12 <u>12-28-83, Amendment 79 to DPR-24, Amendment 84 to DPR-27</u>

These amendments added an additional reporting requirement to the Technical Specifications to report all challenges to the pressurizer power-operated relief valves and pressurizer safety valves in the annual report.

3.1.13 12-29-83, Amendment 80 to DPR-24, Amendment 85 to DPR-27

These revised the Technical Specifications surveillance requirements to include monthly testing of the automatic logic circuitry for the auxiliary feedwater pumps.

3.1.14 12-30-83, Amendment 81 to DPR-24

This amendment authorized Point Beach, Unit 1, reactor operations at either 2250 or 2000 psia after return to power following steam generator replacement.

3.2 Facility or Procedure Changes Requiring Nuclear Regulatory Commission Approval

There were no plant modifications or procedure changes during 1984 beyond those authorized with license ammendments as noted previously, which required Nuclear Regulatory Commission approval.

3.3 Tests or Experiments Requiring Nuclear Regulatory Commission Approval

There were no tests or experiments at Point Beach Nuclear Plant in 1984 which required Nuclear Regulatory Commission approval.

3.4 Design Changes

The following design changes were completed during 1984:

3.4.1 E-200 (Unit 1) & E-201 (Unit 2), Main Steam Stop Valve Solenoid Valves. The Lawrence solenoid valves in the main steam stop valve cabinets were replaced with ASCO sliding ring collar solenoid valves (four valves power steam line).

Summary of Safety Evaluation: The new solenoid valves provide more reliable latching and should eliminate the trips frequently experienced after testing.

3.4.2 IC-106 (Unit 1) & IC-107 (Unit 2), Feedwater Heater 4A&B Level Control. These modifications permit alarming of high water level in the feedwater heater but delays the BTV closure to assist in clearing the high level alarm.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.3 IC-133 (Common), Water Treatment. This modification changed the control scheme to allow the lime feed pump to run continuously with flow integrator controlling the lime feed auger only. This will reduce the rate of lime build-up in pumps and lines.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.4 IC-183 (Unit 1), Air Ejector Instrumentation Upgrade. The modification installed a Hastings linear mass flow meter in the condenser air ejector discharge line. The modification was needed in view of more stringent restrictions on condenser air inleakage as it impacts turbine spindle warranty requirements.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.5 IC-187 (Common), Water Treatment. This modification installed a timer in the anion exchanger regeneration control circuit so that the anion bed can be sufficiently preheated before beginning caustic flow.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.6 IC-261 (Common), Radiation Monitoring System. The modification installed a new compact digital radiation monitoring system with control room readout to replace the existing system. It added new area monitors to address NUREG-0737 requirements. New process monitors were installed to meet RETS requirements. The system also allows better access to data in the technical support center.

Summary of Safety Evaluation: The control logic does not alter the "fail safe" feature of the previous radiation monitoring system for preventing inadvertent releases. Power supply is from the white and yellow instrument buses but isolation transformers are used to protect the instrument buses. Loading of the buses was considered under modification requests E-206/207. Required piping changes were covered by special maintenance procedures.

3.4.7 IC-270 (Unit 2), Bus Cooler Low Flow Alarm. This modification installed a new type flow switch because of repeated failures of the originally installed flow switch.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.8 IC-289 (Unit 2), Pressurizer Safety Valve Direct Position Indication. The modification installed position switches on each pressurizer safety valve to provide direct indication of valve position. Position indication was previously provided by an acoustic monitoring system but was unsatisfactory because it did not meet environmental qualification requirements.

Summary of Safety Evaluation: The system only provides indication; no control functions and is, therefore, not nuclear safety related. The system's importance to safety was recognized and therefore seismic, environmental qualification and single failure criteria applied to the extent practicable.

3.4.9 IC-291 (Unit 2), Wide Range Th & Tc Environmental & Loop Upgrade. The modification is the result of IE Bulletin No. 79-01B and Regulatory Guide 1.97 and replaced the reactor coolant system hot and cold leg loop RTD's along with their associated wiring and electronics. The RTD outputs are processed by the new Foxboro Spec 200 racks.

Summary of Safety Evaluation: Not nuclear safety related but important to safety due to indication provided and signal input to the subcooling meter.

3.4.10 IC-303 (Unit 1) & IC-304 (Unit 2), Main Feedwater. These modifications upgraded the existing steam generator wide range level transmitters and added redundant wide range level transmitters to each steam generator. This was done to meet regulatory commitments.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.11 IC-313 (Unit 2), Incore Thermocouple Upgrade. The modifications replaced reactor vessel incore thermocouple connectors with qualified connectors. It split thermocouples into separate redundant white and yellow channels and routed them to new penetrations using qualified cable, thus bypassing non-qualified junction boxes. The cable from the penetrations was routed to the computer room. Completion of this modification satisfies NUREG-0737, Item II.F.2, which requires that incore thermocouples be used for redundant safety grade subcooling meters as an indication of adequate core cooling.

Summary of Safety Evaluation: The upgraded thermocouple system did not involve changes to the existing thermocouples or penetrations in the reactor head. Therefore, the modifications do not affect the reactor coolant system pressure boundary. The system provides indication only and therefore is not nuclear safety related, although its importance to safety in monitoring core conditions is recognized.

3.4.12 IC-337 (Unit 2), Main Control Boards. The second portion of the main control board modifications, those for 1(2)C03, were installed.

Summary of Safety Evaluation: This modification consolidates the numerous control board changes required by additional instrumentation being installed per NUREG-0737 recommendations. All work on the main control boards was appropriately controlled and scheduled to minimize potential impact upon safe plant operation.

3.4.13 IC-341 (Unit 1), Turbine Generator EH. The modification installed a Flexitest switch for the EH kilowatt transducer to allow calibration and testing without lifting the leads.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.14 M-209 (Unit 2), CRDM Drive Rod Chamfer Increase. This modification request is identical to M-182, performed on Unit 1. Problems have been experienced in other plants during reactor vessel head lowering, with the reactor vessel head tending to hang up on the control rod drive shafts. The mount of the interference under certain conditions was determined to be between 0.030 and 0.041 inches although drawing tolerances could conceivably give a "stack up" of up to 0.070 inches. Therefore, the lead-in chamfer at the top end of the rod drive shaft was enlarged to avoid the potential for interference with CRDM blocks during reactor vessel head installation. All but one drive shaft had been chamfered several years ago. The last one was chamfered during the 1984 refueling outage when the driveshafts were removed for flexure pin inspection.

Summary of Safety Evaluation: The safety evaluation determined that the increased chamfer materially assists in giving the control rods an easier lead into the reactor vessel head. The function of the control rods is in no way affected by the increased chamfer.

3.4.15 M-623 (Unit 1) & M-624 (Unit 2), Steam-Driven Auxiliary Feed Pump Bearing Cooling. The modifications installed valves and crossconnects to the feedwater system to enable auxiliary feed pump bearing cooling in the case of a total loss of AC power, including the emergency Diesel generators.

Summary of Safety Evaluation: This modification gives added redundancy to the cooling water supply to the turbine-driven auxiliary feed pump bearings.

3.4.16 M-662 (Unit 2), Reactor Coolant System Vent System. The modification installed a system which provides a means to vent the reactor vessel or pressurizer to the pressurizer relief tank or containment from the control room. The modification was required by NUREG-0737.

Summary of Safety Evaluation: System design and installation meets or exceeds original plant criteria, and therefore, the integrity of the reactor coolant system is maintained. Orifice couplings provide a class break and flow restriction between the reactor coolant system and downstream piping. Flow paths from the pressurizer and reactor head to the pressurizer relief tank and containment are provided with in-series solenoid-operated isolation valves which fail closed upon loss of power.

3.4.17 M-663 (Unit 1), Primary Sampling. The modification installed a manual isolation valve, accessible during power operation on the hot leg sample line just upstream and adjacent to 1(2)AOV-955.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.18 M-671 (Unit 1) & M-672 (Unit 2), Steam Generator Sampling Isolation Valves. These modifications greater reduced the change of service water backflow into the steam generator sample lines during shutdown. The modifications installed additional isolation valves to ensure service water backleakage would be directed to the sample sink.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.19 M-677 (Common), Technical Support Center. The modification constructed an on-site structure in accordance with NUREG-0578 requirements and also provided additional office space for plant personnel.

Summary of Safety Evaluation: Not nuclear safety related, however, radiation atmospheric control and equipment redundancy provisions will be incorporated into the design.

3.4.20 M-681 (Unit 1), CVCS. The modification installed a containment isolation valve with "T" signal trip in the RCP seal water return line inside containment. The valve is located downstream of relief valve 1-314.

Summary of Safety Evaluation: The new valve provides additional assurance of containment isolation. The valve will fail closed on loss of air or electric power, and it is seismically qualified.

3.4.21 M-683 (Unit 1), CVCS. The modification installed a containment isolation valve on the common letdown line downstream of the orifice block valves inside containment. The isolation valves are equipped with a "T" signal trip.

Summary of Safety Evaluation: The new valve provides added assurance of containment isolation. The valve will fail closed on loss of air or electrical power and is seismically qualified.

3.4.22 M-728 (Common), Hydrazine Addition. The modification replaced the existing common hydrazine pump (P78C) with a Milton Roy controlled volume pump. This was done to treat the feedwater more evenly and thereby reduce feedwater system corrosion.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.23 M-748 (Common), Secondary Sampling. This modification set up a lab preparation table, including sink and 120 V AC connections, to improve response time to changes in feedwater chemistry.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.24 M-788 (Common), Reactor Makeup Water Valving. The modification installed a new in-series manual isolation valve in the reactor makeup water supply header to the boric acid transfer pumps to prevent dilution of a boric acid storage tank if any of the existing valves 1(2)-336A&B leak through.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.25 M-793 (Common), Warehouse No. 3. This modification constructed a new warehouse (~8000 ft^2) to provide a suitable controlled environment for storage.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.26 M-819 (Unit 1), Primary Sampling Test Connection, Containment Isolation Valves. The modification added test connections for AOV-951, 953 and 955 and the corresponding root isolation valve for each.

Summary of Safety Evaluation: Tubing and valve installation meets or exceeds original system requirements. System integrity is not degraded. The modification permits leak testing of the valves in accordance with Appendix J of 10 CFR 50.

3.4.27 M-825 (Common), Ready Stores Heater. This modification installed a heater in the Ready Stores building to meet QA storage requirements.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.28 82-006 (Common), New Radwaste Cryogenic System. This modification made permanent the jumpers which were installed to remove CR-7 from service. This radiation monitor is not used so the change was made permanent.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.29 82-016 (Unit 2), Main Turbine. The modification installed an intermediate latching relay in the 20-ET solenoid valve circuit to provide redundancy in the turbine tripping logic to protect against a failure of the 20-AST solenoid coil or a failure of the OSTR by holding the 20-ET energized until the intermediate relay is reset.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.30 82-021 (Unit 2), Bearing Temperature Recorders (2TR-2000). This modification replaced the existing recorders with new recorders which have essentially instantaneous alarming capabilities and improved readability.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.31 82-028 (Unit 1), Safety Injection System. The modification installed manual operators on 1-897A&B actuators.

Summary of Safety Evaluation: The manual valve operators provide positive means to restore mini-recirc flow for the safety injection pumps and have been evaluated to verify that they do not degrade the integrity of the safety injection system. The manual operators are administratively controlled to ensure auto closure is not prevented upon opening of valve 851A&B.

3.4.32 82-031 (Unit 1) & 82-32 (Unit 2), Containment Ventilation. The modifications interlock the refueling surface exhaust fan (W5A) to the purge exhaust fans such that both purge exhaust fans must be operating in order for W5A to operate. Previously, only one purge exhaust fan had to be operating in order for W5A to operate. The capacity of W5A exceeds that of a single purge exhaust fan.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.33 82-046 (Unit 1) & 82-047 (Unit 2), Primary Sampling Valve Control Panel. The modifications moved the reactor coolant sample valve control panels from outside pipeway Nos. 1 & 4, respectively, to the vicinity of the primary sample rooms on PAB El. 26'. The new locations were selected because they were accessible and shielded from high radiation areas even during assumed post-accident situations.

Summary of Safety Evaluation: Relocation of the sample valve control panels does not alter the control or function of the valves. Design and installation complied with appropriate QA requirements and functional testing verified valve operability after location of the control panels.

3.4.34 82-046-01 (Unit 1) & 82-047-01 (Unit 2), Primary Sampling. Addendum 1 to the modifications removed the containment isolation signal for valves 951, 953 & 955 from the "B" train safeguards logic. The DC power supply for these valves is the "A" train.

Summary of Safety Evaluation: The change eliminated the "B" train containment isolation signal from the primary sampling system isolation valves 951, 953 & 955. The "A" train safeguards logic alone provides the isolation signal to close these valves. Valves 966A, 966B & 966C are redundant with the above valves and receive containment isolation signals from both "A" & "B" trains. This satisfies the single failure criteria of Technical Specification 15.5.2.b.2. No Technical Specification change will be required. A failure of the new (modified) circuitry would not lead to an unanalyzed accident as the worst case would be the failure of all three valves (951, 953 & 955) to close; valves 966A, 966B & 966C would be unaffected and would close to maintain containment integrity. The probability of a failure to maintain containment integrity is not increased since both sets of valves operate independently of one another. Installation was controlled by procedure to ensure proper operation.

3.4.35 82-056 (Common), Gatehouse Expansion. This modification expanded the south gatehouse by approximately 4,000 ft^2 to alleviate the crowded conditions.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.36 82-061 (Common), Underground Ducting. The modification extended the existing duct bank from the No. 3 warehouse to a new manhole 55' north of the plant perimeter fence. Cable was installed for 4 KV service from the warehouse switcher 2A07 to a new switcher. The modification was needed for steam generator replacement activities and for supply power to the new north meteorological tower.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.37 82-065 (Common), Fire Door Surveillance. The modification placed all remaining safe shutdown fire doors, on the plant security system. These doors were not placed in access because this would remove them from surveillance. All other fire doors on the security system were secured or software changes made to ensure continuous surveillance.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.38 82-070 (Unit 2), Fuel Handling. The modification marked the fuel-in-core zone position of the fuel manipulator hoist speed control switch to provide control of hoist speed when fuel is being raised or lowered within the core.

Summary of Safety Evaluation: Not required.

3.4.39 82-076 (Unit 1) & 82-077 (Unit 2), Main Steam System. The modification replaced the tube bundles in all moisture separator reheaters in each unit with a new design.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.40 82-078 (Unit 1) & 82-079 (Unit 2), CVCS Letdown. The modifications provided for automatic closure of AOV-200A, B&C whenever either letdown containment isolation valve 371 or 371A is not fully open by installation of a relay in the position indication circuitry of AOV-371&371A.

Summary of Safety Evaluation: The modifications minimize unnecessary lifting of letdown line relief valve RF-203 when a containment isolation trip signal is present. This minimizes loss of reactor coolant system pressure via the 2" line as well as minimizing the loss of water. The 600 psi piping between the letdown line orifice isolation valves and the containment isolation valves also incur less cycling loading. The circuitry modifications do not affect any containment isolation functions.

3.4.41 82-084 (Unit 2), Feedwater System. The modification replaced the No. 4 feedwater heaters because extensive corrosion of the tubes has reduced tube wall thickness. The new heaters have Type 304 stainless tubes.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.42 82-102 (Common), Emergency Diesels. The modifications replaced ¼" copper diesel sensing lines with stainless steel tubing.

Summary of Safety Evaluation: Replacing the sensing lines does not affect the function or control of the emergency Diesel generator. Reliability was improved by the higher fatigue resistance of stainless tubing over copper tubing.

3.4.43 82-110 (Unit 1), Reactor Coolant System Pipe Hanger. The modification relocated hanger RC-1 from the east to the west side of the seismic support to which its support beam is attached in order to eliminate interference problems experienced in accessing the "B" steam generator north handhole during sludge lancing.

Summary of Safety Evaluation: The effect on the reactor coolant system piping of relocating hanger RC-1 has been evaluated by Bechtel and found to be acceptable. All other design and installation requirements remained unchanged.

3.4.44 82-114 (Unit 1), Pressurizer Steam Space AOV's. The modification replaced the pressurizer steam space sample air-operated valves 951 and 966 with manual valves. The AOV's were used to replace leaking hot leg and pressurizer liquid sample AOV's 955 and 953.

Summary of Safety Evaluation: The manual valves meet or exceed primary sample system specifications. The valves will be left closed during operation because pressurizer steam space samples are not required.

3.4.45 82-115 (Unit 2), Seal Oil Package. This modification crossconnected the air side seal oil duplex filter drain lines so that the off-line filter can be pressurized slowly. This prevents the potential sampling of seal oil onto hot piping which would create a fire hazard.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.46 82-119 (Unit 1), Steam Generator Vents. This modification moved isolation valves 211 and 212 to allow for safe access and operation from the missile shield.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.47 82-122 (Unit 1) & 82-123 (Unit 2), Main Feed. These modifications installed an indicator light next to the feedwater control valve bypass valve controller that would indicate if the bypass valve trip at steam generator high level had been actuated.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.48 82-124 (Unit 1) & 82-125 (Unit 2), Building, Structures. The modifications removed containment penetration 1(2)P56 valves and installed a blank flange outside containment. The penetrations were not being used.

Summary of Safety Evaluation: The change, via either a welded or gasketed seal, does not involve an unreviewed safety issue. The change to the penetrations results in a configuration that is equal to or better than the sealing and seismic capabilities of the previous configuration. The materials and installation practices, used in incorporating the change are also equal to or better than existing equipment. A passive containment boundary is provided which is fully testable.

3.4.49 83-004 (Unit 1), Primary Sampling. The modifications replaced 1-953 and 1&2-955 with valves of higher quality to minimize leakage.

Summary of Safety Evaluation: The new valves function the same as the previous valves and meet or exceed original system design and installation requirements. Seismic evaluations were performed to ensure seismic qualification is maintained.

3.4.50 83-009 (Common), Lighting. This modification installed new lighting fixtures above the aisles between the shelving in warehouse No. 3.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.51 83-010 (Unit 1) & 83-011 (Unit 2), Instrument Air/Sampling System. The modifications installed environmentally and seismically qualified solenoid valves, limit switches, conductor seal assemblies and cable splices on 2CV-3047, 2CV-3048 (instrument air isolation valves) and 2SV-959 (residual heat removal sample valve).

Summary of Safety Evaluation: Components were replaced on a part-for-part basis with qualified components and do not change the operation and control of the valves. Work was performed by procedure and the valves were functionally tested after work was completed to ensure their operation.

3.4.52 83-025 (Unit 1) & 83-026 (Unit 2), Relaying. The modification replaced the existing loss of voltage relays with new relays having a fixed time response.

Summary of Safety Evaluation: The new relays perform the same function as the existing relays but can be set to ensure that no 480 V safeguards load operates at less than 75% of rated voltage for more than approximately 0.4 seconds. The new relays do not change state on loss of DC and therefore do not increase the probability of inadvertent actuation. Installation was controlled by special maintenance procedure. The time delay setting of the new relays is the same or less than the response time of the previous relays for all voltage conditions.

3.4.53 83-028 (Unit 1), CVCS Charging Pumps. This modification rerouted the charging pump leakoffs to the existing pipe for leakoff from the head tray. This was done to alleviate the buildup of boric acid around the pump block which resulted in high beta radiation levels.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.54 83-031 (Unit 1), 83-32 (Unit 2) & 83-33 (Common), Air Ejector. These modifications installed permanent sample systems in the condenser air ejector discharge lines.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.55 83-036 (Common), Perimeter Intrusion Detection. This modification installed a four-wire balanced phase system and a microwave detection unit at the personnel gates leading to switchyard and potable water pumphouse. This was done to increase detectability so that site penetration cannot be made without alarm.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.56 83-52 (Unit 2), Fuel Handling. The modifications procured new fuel manipulator grippers with design features to positively prevent gripper engagement or disengagement unless the manipulator mast is properly positioned upon the fuel assembly.

Summary of Safety Evaluation: The new gripper assembly prevents engagement unless properly positioned on a fuel assembly, thus reducing the potential for a fuel handling accident. The design material and craftsmanship is equal to, or better than, the original equipment.

3.4.57 83-058 (Common), Buildings & Facilities. The modification installed a containment access building, equipped with utilities, for use during the Unit 1 steam generator replacement outage. The modification is temporary in nature and utilities will be capped below grade and pavement repaired following conclusion of the project.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.58 83-063 (Unit 2), Polar Crane. The modification installed dual upper limit switches to deenergize the main power supply and hoist drive motor.

Summary of Safety Evaluation: This modification removed the cable actuated limit switches from the hoist motor circuits, for both the main and auxiliary hoists, and placed them in the main contactor circuits. This meets the requirements of NUREG-0554 which necessitates that independent mechanical means be used to deenergize the main power and hoist drives separately. The geared limit switches continue to be used in the hoist motor circuits. The changes ensure positive independent hoist shutdown in the event of a potential "two-blocking" event. The modification change was controlled by a special maintenance procedure. No changes to Technical Specifications were required.

3.4.59 83-079 (Unit 1), Switchyard. The modification added instantaneous overcurrent relays connected to separate current transformers in the ground grid. The overcurrent relays were connected to initiate the 345 KV No. 2 bus differential lockout. The modification detects a failed 345 KV circuit and initiates fault clearing.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.60 83-085 (Unit 2), Cavity Access Ladder. The removable top section of the cavity access ladder and cage was modified to allow easier alignment with lower section during installation .

Summary of Safety Evaluation: Not nuclear safety related.

3.4.61 83-089 (Unit 2), Extraction Steam. This modification moved steam trap line ST-2381 just upstream of MS-72 in order to reduce the possibility of water hammer.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.62 83-095 (Common), Potable Water. This modification eliminated all six crossconnections of potable to non-potable water. This was done to prevent contamination of the potable water system and to satisfy Wisconsin State Code 62.14.1.c.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.63 83-097 (Unit 1), Temporary Transformer Installation. In order to provide electric power needed for steam generator replacement activities, one 300 KVA transformer was connected to the existing 1P1A 4 KV cables outside of containment, and three 1500 KVA transformers were connected to the 1P1B cables inside containment. The COM-5 overcurrent relays on 1A52-04 and 1A52-14 were replaced by COM-9 overcurrent relays with appropriate overcurrent elements. Following steam generator replacement, restoration to the original condition was performed.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.64 83-104 (Common), Auxiliary Feedwater System. The modification provides automatic operability to the motor-driven auxiliary feedwater pump discharge motor-operated valves so the affected unit's valves open on an AFWS automatic actuation signal and the unaffected unit's MOV's shut. Control is such that the valves can be shut if the steam generators are required or desired to be placed in a low-low or drained condition.

Summary of Safety Evaluation: This modification uses very similar logic and circuitry as presently used for the motor-driven AFW pump for the automatic startup of the same motor-driven AFW pump and also the automatic operation of each pump's discharge MOV's. Train separation is maintained throughout the schemes of the modification so that single failure criteria of safety grade signals is maintained. In each train of automatic logic (for pump startup and automatic valve operation) the modification divides unit signals so that an automatic startup signal from each unit will energize its respective follower relay (a QA-qualified NBFD relay). The follower relay will then allow for automatic pump startup and required valve cycling (open valves to affected unit, close valves to unaffected unit) to ensure automatic delivery of AFW system flow without operator action.

A minor change was made to the present SI/safeguards sequence circuitry configuration to ensure that an SI signal in one unit would not lock out a steam generator low-low level or trip of both main feedwater pumps signal in the other unit for more than 10 seconds (time of present safeguards sequence for AFW pump startup). This ensures that if AFW flow is required to be sent to both units simultaneously, it will be delivered automatically to both units. It will then be the operator's responsibility to direct AFW flow by taking manual control of the discharge valve control switches. Also, automatic valve operation will still be assured even when the operator manually starts the motor-driven AFW pump.

3.4.65 83-104-01 (Common), Auxiliary Feedwater System. Addendum 1 to the modification removed steam generator low level relays 1(2)LCB and 1(2)LCA and taped off associated leads. These relays were installed during original plant construction and were not used. All wiring to the relays has been disconnected and removed.

Summary of Safety Evaluation: The only effect this relay removal had upon plant operation concerned disconnection of lead 1B12C-P2, which goes between the P38A control switch (contact B12) and relay 2LBA (terminal 9), and lead 2B31C-P2 between the P38B control switch (contact B12) and relay 2LCA (terminal 10). If one of these leads was grounded, such grounding could prevent an automatic start of P38A or B. Therefore, direct supervisory observation was required to ensure that the leads were properly lifted at the relay end and taped to provide proper insulation until modification 83-104 was installed, at which time the leads were disconnected.

3.4.66 83-104-02 (Common), Auxiliary Feedwater. Addendum No. 2 to the modification installed a white light and relay in each auxiliary feedwater train for visual and audible indication/alarm on loss of power to the relay control circuit for automatic operation of the electric-driven feed pumps and valves. The alarm is "AFWS Disabled U1/U2."

Summary of Safety Evaluation: Addition of the power supply monitoring relays enhances safety in that the operator can be assured of automatic auxiliary feedwater system operability, provided that the Unit 1 or Unit 2 auxiliary feedwater system disabled alarm is not received. The failure mode of the relay is such that if the relay fails, a set of normally closed contacts on the relay will make up to bring in the annunciator on C01. The white light aids in operator identification of the problem should he receive a Unit 1 or Unit 2 auxiliary feedwater system disabled alarm.

3.4.67 83-105 (Unit 1), New Steam Generators. The modification replaced the lower assemblies and refurbished the existing upper assemblies including replacement of the primary moisture separators, main steam and other piping as required.

Summary of Safety Evaluation: See NRC safety evaluation report, Docket No. 50-266, dated July 15, 1983.

3.4.68 83-107 (Unit 1), Steam Generator Level Instrumentation. The modification altered the piping arrangement of the two wide range water level transmitters to provide additional redundancy.

Summary of Safety Evaluation: Design and installation requirements meet or exceed those for the existing wide range level piping. Connecting a wide range level transmitter to a narrow range level transmitter needs to be fully evaluated for possible effects on the safeguards (narrow range) channel. The addition of the second wide range level transmitter to each steam generator was approved with the exception that it should not be tied in to a narrow range sensing line.

3.4.69 83-107 (Unit 1), Steam Generator Level Instrumentation. Addendum 1 to the modification increased the number of valves in the wide range lower tap line from 1 to 2 to be consistent with existing construction. It also changed the location of the tee connection for the upper taps on "A" steam generator from LT-461 and LT-460A to LT-463 and LT-460B; this was done to remove the connection from LT-461, the control channel. The upper fluid line tee connection between one of the narrow range level transmitters and one of the wide range level transmitters was retained in this addendum. This configuration has the advantage that it allows the two wide range level transmitters used for post-accident monitoring to be independent of one another as required by Regulatory Guide 1.97, but has the disadvantage of coupling one of the wide range transmitters to a narrow range transmitter used for reactor protection. Connections to narrow range lines are outside the biological shield.

Summary of Safety Evaluation: This addendum removed the originally proposed crossconnection of a wide range channel on the "A" steam generator to the controlling narrow range channel. A crossconnection was made between a wide and a noncontrolling narrow range channel on each steam generator. Failure of a wide range transmitter and/or its associated electronics will not affect the narrow range channel. Piping and tubing design meets or exceeds existing design to ensure the integrity of the system is not degraded.

3.4.70 83-108 (Unit 1), Reactor Coolant System. The modification added calcium silicate box insulation jacketed with stainless steel on the loop seal piping between the pressurizer and the safety valves.

Summary of Safety Evaluation: Insulating the loop seals helps to ensure maintaining the integrity of the reactor coolant system by reducing the dynamic loads on the safety valves and piping when the valves operate. The increased temperature of the loop seal piping is within its design limits. The modification is consistent with EPRI studies.

3.4.71 83-109 (Unit 1), Reactor Coolant System. The modification upgraded pressurizer safety valve discharge piping by deleting some existing supports and adding new supports.

Summary of Safety Evaluation: Design of the new supports is based upon analyses to ensure stress limits are not exceeded. The new supports provide additional assurance that the integrity of the subject piping will not be degraded by relief valve operation. Per Technical Specification 15.3.15.5, there is no requirement to inform NRC prior to installation.

3.4.72 83-110 (Unit 1) & 83-111 (Unit 2), Feedwater Control & Bypass. The modification installed environmentally and seismically qualified solenoid valves, limit switches, electrical conductor seal assemblies, and cable splices on the main feedwater control and bypass valves (1&2CV-476, 481, 466 & 480).

Summary of Safety Evaluation: The function and operation of the main feed control bypass valves is not affected by this modification. All replacements will be part-for-part with qualified components.

3.4.73 83-11-01 (Unit 2), Feedwater System. Addendum No. 1 to the modification changed out existing ASCO solenoid-operated valves 2SV-466C, 2SV-476C, 2SV-480 and 2SV-481 to Model L206-381-6F and added new SOV's 2SV-466D and 2SF-476D to operate in parallel with 2SV-466C and 2SV-476C, respectively.

Summary of Safety Evaluation: Replacement of the original solenoid-operated valves with those specified in this addendum are considered satisfactory since the new valves are fully qualified (per IEEE-323, IEEE-344, NUREG-0588 and IE Bulletin No. 79-01B) and are equivalent or better than the previous configurations. The additional air dump SOV's ensure rapid closure (≤15 seconds) of the main feed regulating valves on a safety injection signal (or high-high steam generator level) to show adequate margin to DNB for an MSLB accident. Operability of all valves was verified by operational testing. The new SOV's are normally energized and therefore failure of a solenoid coil or power supply will result in the conservative action of closure of the main feed regulating valves. Installation was controlled by written procedure.

3.4.74 83-112 (Unit 1), Temporary Radwaste Building. This modification erected a temporary radwaste handling building between the CACE and the hot maintenance shop for the segregation and handling of clean and contaminated waste generated during the steam generator replacement project.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.75 83-119 (Unit 2), Safety Injection System. The modification installed manual operators on 2-897A&B actuators.

Summary of Safety Evaluation: The manual valve operators provide positive means to restore mini-recirc flow for the safety injection pumps and have been evaluated to verify that they do not degrade the integrity of the safety injection system. The manual operators are administratively controlled to ensure auto closure is not prevented upon opening of valves 851A&B.

3.4.76 83-123 (Unit 2), Reactor Trip Breakers. The modification installed a shunt trip relay in the reactor trip breaker, in parallel with the undervoltage trip relay. Actuation of the STR is in addition, and as backup for, actuation of the undervoltage trip attachment. The modification was necessary to meet the requirements of Generic Letter 83-28.

Summary of Safety Evaluation: A plant-specific unreviewed safety question was determined to exist concerning this modification. The safety questions were described in the attachment to the August 10, 1983, SER, submitted to the Westinghouse Owner's Group. Satisfactory draft responses to the questions are contained in a March 30, 1984, memo (Katers to Zach) as modified on May 20, 1984. The responses resolved safety questions raised and contain proposed Technical Specification changes which were reviewed by the NRC. Installation of the modification was controlled via special maintenance procedure.

3.4.77 83-124 (Unit 1), Reactor Coolant System. The modification removed a 6" check valve with a seal welded bonnet from the capped "B" loop hot leg safety injection connection. It further removed loop decon valves 1-543 & 1-544 and capped the connections. The valves had been sources of primary system leakage to containment.

Summary of Safety Evaluation: The hot leg safety injection connections perform no function and thus their removal will improve reactor coolant system integrity. Design and installation will meet or exceed original system requirements.

3.4.78 83-125 (Common), Service Water/Auxiliary Coolant System. The modification installed a 2" diameter tie line between the systems to facilitate discharge of the refueling water storage tanks and boric acid storage tanks. The line was installed between 2" diaphragm valve 2SF-819A and the 2" blank flanged stub connection on the service water supply line to HX-13B.

Summary of Safety Evaluation: The ACS/SW crossconnect is a temporary installation. A seismic evaluation of the service water connection and a hydrostatic test was performed to ensure the integrity of the two systems is not degraded. Use of the crossconnect is controlled by procedure. Corrosion effects on the carbon steel piping will be negligible due to the short time duration, temperature and flushing action of the service water flow.

3.4.79 83-128 (Unit 1), Extraction Steam Piping. The modification replaced all extraction steam system fittings and piping to the Nos. 4 & 5 feedwater heaters.

Summary of Safety Evaluation: The extraction piping serves no safety-related function and its failure cannot affect safety-related equipment because none is located in the adjacent areas. An engineering analysis has been performed which verifies acceptability of the stainless piping and ensures that the integrity of the system is not degraded.

3.4.80 83-129 (Unit 2), Extraction Steam. The modification replaced the existing carbon steel extraction piping to the Nos. 4 & 5 feedwater heaters with stainless steel piping.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.81 83-134 (Unit 1), Containment Polar Crane. The modification attached braces to the crane bridge girders for the center post modifications needed for steam generator replacement. The remaining braces do not alter the operating characteristics or load rating of the crane.

Summary of Safety Evaluation: The structural integrity of the crane was verified prior to returning the crane to normal service.

3.4.82 83-138 (Unit 2), Turbine Lube Oil. This modification installed a turbine bearing oil lift system on the low pressure bearings to prevent slip sticking at low revolutions per minute.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.83 83-139 (Unit 1) & 83-140 (Unit 2), Loose Parts Monitoring. The modifications provided a loose parts monitoring system in the primary and secondary cooling systems, which will detect loose parts which could potentially cause damage to the steam generators. The equipment included sensors on the bottom of the reactor vessel and on each steam generator. Of the 10 accelerometers in each unit, 6 will be active and 4 will be passive. The active channels provide continuous impact monitoring while the passive channels consist of in-containment equipment and cabling to the control cabinet which is located in the computer room. The system is not seismically or environmentally qualified. The system does, however, include provisions for self-testing and will alarm upon equipment failure as well as providing alarming upon detection of loose parts.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.84 83-144 (Common), Cranes & Hoists. The modification provided an I-beam trolley and chain hoist arrangement in the G01 emergency Diesel generator room. It enables Maintenance personnel to more efficiently remove the cylinder banks of the emergency Diesel generator so the seal between the banks and the crankcase can be inspected.

Summary of Safety Evaluation: An evaluation of the lifting beams was performed to ensure they will remain intact during a seismic event. Actual use of the beams is only for emergency Diesel generator maintenance.

3.4.85 83-156 (Unit 1), Component Cooling. This modification installed a vent with an isolation in the vicinity of the component cooling pump suction high point. Previously, the return header could not be adequately filled and vented causing air binding of both component cooling pumps during fill and vent operations.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.86 83-164 (Unit 2), Turbograph Recorder Changeout. This modification replaced the obsolete Honeywell Electronic 16 recorders with Speedomax 2500 recorders.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.87 83-171 (Unit 1), Facade Building. This modification installed a handrail, lighting and a permanent access ladder to the facade stairwell roof for personnel safety concerns.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.88 83-181 (Unit 1), Secondary Sampling. This modification removed and capped existing sample points on the bottom of the main feedwater lines and replaced them with sample isolation valves on the side of the main feedwater pipe. This was done to make the sample line less prone to plug up.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.89 83-187 (Common), Condenser Air Removal System. This modification installed a local heater overtemperature indicating light for the condenser air removal filter, F30.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.90 84-007 (Unit 1), Plant Distribution. The modification replaced the old overcurrent tripping devices with LSI Amptector overcurrent tripping devices on the rod drive motor breakers. The tripping device is completely separate from the shunt trip coil. No separate power source was needed.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.91 84-008 (Unit 2) & 84-009 (Unit 1), 4160 V Protective Relaying. This modification replaced existing KF relays with KF relays manufactured after 1975. They were replaced to prevent possible failure of the relays at high temperature conditions.

Summary of Safety Evaluation: Relay replacement was with an improved design relay by the same manufacturer and which have been in use since 976. No other changes to the function or design of the underfrequency trip circuit were involved. Installation was controlled by procedure and standard electrical tests of the replacement relays were performed.

3.4.92 84-010 (Unit 1), Service Water. The modification removed first-off isolation valves SW-614 and SW-604 and installed a screwed pipe cap at the header, eliminating a stagnant line freeze potential.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.93 84-012 (Unit 1), Service Water. This modification removed the spacing line between service water and steam generator blowdown tank. A threaded pipe plug was installed on one end and a socket weld pipe cap on the other end.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.94 84-013 (Unit 1), New Steam Generators. The modification welded a nut to ¼" plate with 4 tabs extending between the volt heads and fastened to the edge of the steam generator manway strongback with 4, ½" bolts.

Summary of Safety Evaluation: The 4, ½" bolt holes added to each manway cover are located in a low stress area of the cover similar to the existing 1" bolt hole provided for an eyebolt attachment. Due to the much smaller size and location of these additional holes, the modification has negligible effect on the strength of the manway cover. An analysis was performed by Westinghouse to verify the acceptability of the modification.

3.4.95 84-014 (Unit 1), Main Steam System. The modification installed a welded diaphragm plate in lieu of the existing gasket for the sacrificial flange.

Summary of Safety Evaluation: The diaphragm plate provides only a sealing function. Materials and welding requirements have been specified which are compatible with main steam piping to ensure that the integrity of the flanged connection is not degraded.

3.4.96 84-018 (Common), Technical Support Center. This modification removed a section of the counter top and shelving from the northwest wall of Room 108 in the technical support center. This space was needed for a Canberra S-8100 multichannel analyzer and a Digital 11/05 computer to improve Emergency Plan capabilities.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.97 84-028 (Unit 2), Reactor Coolant System. The modification ground off the nuts on the steam generator manway covers and drilled four ½" UNC x 1½" maximum deep holes into the cover sides to permit attaching a nut/bracket assembly, as was performed on the new Unit 1 manway covers (reference modification request 84-13).

Summary of Safety Evaluation: The four ½" bolt holes added to each manway cover are located in a low stress area of the cover similar to the existing 1" bolt hole provided for an eyebolt attachment. Due to the much smaller size and location of these additional holes, the modification had negligible effects on the strength of the manway cover. Westinghouse concluded for modification 84-13 that the configuration complies with all structural requirements of ASME Section III. Fatigue in the vicinity of the bolt holes is acceptable. The modification was performed in accordance with Code requirements. These requirements, heat treatment and NDT will deter possible adverse effects.

3.4.98 84-029 (Unit 1) Turbine Trip Block. This modification installed a bushing on the trip seat to reduce the trip seat bore so that the turbine trip will reset.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.99 84-033 (Common), Station Batteries. This modification increased the thickness of the support straps for the station batteries from 3/16" to 3/8" to improve the seismic rating of the support racks.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.100 84-035 (Unit 1), Service Water. This modification installed valves closer to the oil coolers on 1P28A&B, making them more accessible for maintenance and operation.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.101 84-041 (Unit 1), Feedwater Vents, Reliefs, Drains. This modification installed sample points in the 3A, 4A & 5A feedwater heater vent pipes in order to measure ammonia concentrations.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.102 84-042 (Unit 2), Reactor Coolant System. This modification reinstalled pipe snubber 2HS-M75 until Technical Specifications are changed in order to avoid unit shutdown per NRC interpretation of Technical Specification Limiting Conditions for Operation.

Summary of Safety Evaluation: The reinstallation of snubber supports will cause some minor changes in pipe stresses but will still be within allowable stress limits.

3.4.103 84-047 (Unit 1), AC & Instrument Bus. The modification changed out the supply breakers to instrument distribution panels 1Y11, 1Y31 and 1Y41 from 5 to 20 amps.

Summary of Safety Evaluation: A single fault on 1Y11, 1Y31 or 1Y41 would cause the respective supply breaker on 1Y01, 1Y03 or 1Y04 to trip since the supply breakers on the YO panels are rated equal to or lower than their feeder breakers on 1Y11, 1Y31 or 1Y41. This could result in loss of power to some safeguards instrumentation due to improperly coordinated tripping protection. Since no actual change in loading was made on 1Y01, 1Y03 or 1Y04 buses, it is important that proper tripping protection be provided to prevent an inadvertent loss of power to safeguards instrumentation. Also, since proper tripping protection is provided on 1Y02 and all Unit 2 YO buses by a 20 amp breaker, the use of a 15 or 20 amp QA-qualified breaker on 1Y01, 1Y03 and 1Y04 will provide equivalent tripping protection. The change does not involve an unreviewed safety question and it does not involve a change to the Technical Specifications.

3.4.104 84-050 (Unit 1) & 84-051 (Unit 2), Safety Injection System. The modification installed high point vent line isolation valves on the high head safety injection pump casings.

Summary of Safety Evaluation: Addition of these valves on the pump vents does not affect the safe performance of the safety injection pumps or connecting system. This conclusion is based upon the ability of the added valves to maintain the pressure and temperature requirements of the system and that the valves with their small relative mass increase would not seismically affect the integrity or the safety function of the safety injection pumps.

3.4.105 84-053 (Unit 1), Steam Generator Blowdown. The modification replaced the existing throttle valve downstream of the blowdown heat exchanger with a smaller throttle valve which can be operated in the full open position. The previous valve was oversized which necessitated operation with the valve barely cracked open and with the main steam isolation valve throttled. This caused "chugging" in the system and excessive valve wear and possibly heat exchanger wear/damage.

Summary of Safety Evaluation: The change in valve size has no detrimental effect upon the safe operation of the plant. The capabilities of the blowdown system are not changed detrimentally, however control of the system is improved.

3.4.106 84-055 (Common), Spent Fuel Cask Lifting Yoke. The modification installed an aluminum bar across the fingers on the lifting yoke guide bracket to increase the bearing surface of the fingers and eliminate deformation of the finger corners.

Summary of Safety Evaluation: The alignment guide bracket is not a load bearing component; therefore, this modification does not increase the potential or consequences of dropping the spent fuel shipping cask.

3.4.107 84-069 (Common), Water Treatment. This modification installed a heater and vibrator in the coagulant (alum) feed hopper. This was done to prevent arching in feed hopper, caused by moisture pickup by the coagulant during the warmer, high-humidity months, which stops feed to the raw influent.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.108 84-081 (Common), Service Building Old Maintenance Office. This modification remodeled the room to provide additional HP office space.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.109 84-111 (Unit 2), Residual Heat Removal. The modifications installed tees and Whitey vent valves on the residual heat removal pump seal flushing line to vent the pump seals.

Summary of Safety Evaluation: The fitting and valve installation meets or exceeds original system requirements. System integrity was not degraded. The small additional mass does not affect the seismic qualification of the system.

3.4.110 84-121 (Unit 2), Deaerate Condensate. The condenser inlet pipe was modified to spray the incoming condensate more evenly over the main condenser tube bundle. This was due to reduce the oxygen content of makeup water.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.111 84-135 (Unit 2), Fuel. The modifications will make a transition during future reload cores from 1/3 OFA to an all-OFA core. The gradual transition will increase efficiency of the core by reducing the amount of parasitic material and will reduce fuel cycle costs due to an optimization of the water tc uranium ratio. Major differences are use of 5 intermediate Zircaloy grids for OFA versus Inconel grids for standard fuel and a reduction in fuel rod diameter and guide thimble diameter.

Summary of Safety Evaluation: The Staff accepts the safety and accident analyses performed by Westinghouse. The safety evaluation and Technical Specification changes were filed with the NRC via Change Request No. 87.

3.4.112 84-138 (Common), Waste Disposal. The modification installed a threadolet to the tee above valve WS-38 to permit cleaning of the crud trap above WS-38.

Summary of Safety Evaluation: The modification does not affect the safe operation of the plant; the work was done in accordance with ANSI B31.1.

3.4.113 84-139 (Unit 2), Condensate System. The modifications changed the condensate pump seal supply piping configuration to provide better seal cooling and lubricating water to improve seal life.

Summary of Safety Evaluation: Relocation of condensate pump seal water supply lines has no adverse effect upon the FSAR safety analysis nor the Technical Specifications. This is based upon the fact that the condensate pumps are not essential for plant safety and are not required in any of the safety analyses. New piping and connections are consistent with existing piping and ANSI B31.1-1967.

3.4.114 84-147 (Common), Spent Fuel. The modification elevated the spent fuel cask cover lifting rig 3' from its previous elevation to provide clearance between the lifting rig and the inner head valves located on the top of the head. The permanently mounted valves on the inner head interfere with attachment of the lifting rig to the lifting lugs.

Summary of Safety Evaluation: The inner head lifting rig extension legs are designed to support the weight of the inner closure head (900 pounds) with a safety factor of 3. According to Technical Specification 15.3.8.B.1, one ton shall be the maximum load allowed over either the north or south half of the spent fuel pit when spent fuel which has been subcritical for less than one year is stored in that half of the spent fuel pit. Since the inner head is 900 pounds, it complies with the requirements of 15.3.8.B.1. According to PBNP 3.4.5, "Special Structural Limitations on the Lifting of Heavy Loads," a heavy load is defined as any load greater than 1,750 pounds. The inner closure head is not considered a heavy load by that definition. The lifting rig extension legs, as designed, meet the requirements of Technical Specification 15.3.8.B.1 and PBNP 3.4.5.

3.4.115 84-168 (Common), Safety Injection. This modification installed valve switch covers over SI-834A&B switches on the back of C01 to provide administrative control of switch operation.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.116 84-173 (Common), Battery Rooms. This modification installed cable through battery room walls to allow individual cell charging without using extension cords which prevented closing doors.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.117 84-192 (Unit 2), Main Steam. The modifications changed the main steam nonreturn valve shaft/arm attachments to more securely retain their counterweight arms.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.118 84-206 (Unit 2), Turbine Bearing Thermocouples. This modification installed thermocouples in the four low pressure turbine bearings to permit monitoring bearing metal temperatures instead of only oil temperature.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.119 84-217 (Unit 2), Sampling. This modification allows better drainage of the sample sink. A vent was installed in the sample sink drain line and run to the sampling hood.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.120 84-220 (Common), Spent Fuel Assembly. This modification provided special tools for lifting a spent fuel assembly damaged at the off-site storage facility.

Summary of Safety Evaluation: All lifting equipment was designed for a static load of 1300 pounds with a safety factor of 6. The lifting tool length, 31'9", is not so long as to allow an assembly to be raised to hazardous height. A drop accident is avoided by maintaining a safety factor of 6. The lifting pole assembly was load tested to 6600 pounds and the PBNP spent fuel handling tool was load tested to 2500 pounds. Only one fuel assembly can be moved at a time.

3.4.122 84-223 (Unit 2), Incore Flux Mapping. The modification replaced the existing 36 thimbles with new thimbles having a larger inner diameter (0.213" versus 0.202").

Summary of Safety Evaluation: Replacement of the incore flux thimbles with thimbles of a larger diameter helps to alleviate thimble blockage. The new thimbles meet or exceed original system ratings and do not degrade system integrity. Replacement of the thimbles does not affect a previously reviewed, or create an unreviewed, safety question.

3.4.123 84-225 (Common), Meteorological Monitoring. This modification installed a meteorological monitoring tower to detect the onset of a lake-induced wind pattern different than the weather pattern induced wind.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.124 84-229 (Unit 2), Bypass Manifold RTD Guards. This modification redesigned the bypass manifold RTD guards to allow removal of the guard without removing manifold insulation. The guards were made of stainless steel to prevent corrosion.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.125 84-231 (Unit 2), Reactor Coolant System. The modification ground off the nut welded to the pressurizer manway cover and installed a new bracket as was performed per modification requests 84-13 and 84-28 on the steam generators.

Summary of Safety Evaluation: The four ½" bolt holes being added to the manway cover are located in a low stress area of the cover similar to the existing 1" bolt hole provided for an eyebolt attachment. Due to the much smaller size and location of these additional holes, the modification has negligible effects on the strength of the manway cover. Westinghouse concluded for the Unit 1 steam generators that modification of the manway covers results in a configuration which complies with all structural requirements of ASME Section III. Fatigue in the vicinity of the bolt holes is acceptable. Note there is a material difference between the cover analyzed in Westinghouse report SG 84-03 (SA-533 Grade "A" Class 1 versus SA-302 Grade "B"). However, both of these materials have the same stress allowables; therefore, the results can be translated to this modification. No Technical Specification changes are required.

3.4.126 84-234 (Unit 2), Head Vent Shroud. This modification cut the shroud in the area the head vent pipe penetration so that the shroud segment could be removed for access to the head in the area of the conoseal ports.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.127 84-236 (Unit 2), Fuel. The modifications removed four flexure pins from each of 33 control rod guide tube inserts and replaced these with flexureless inserts.

Summary of Safety Evaluation: The flexureless inserts act as a guide for drive rod travel and also as a flow restrictor to limit flow to the upper head region. The flexureless insert is an improved design by Westinghouse which performs the same function as the original flexure and insert (donut) arrangement. The need for four flexures to hold the insert in place from above the insert is eliminated by use of the flexureless insert. The flexure heads will be removed, but removal is not necessary to permit use of the flexureless inserts. The flexureless insert is held in place by a mechanical spring mechanism which exerts an outward force on four latches against the upper housing plate of the guide tube. The resulting pull-out force for the flexureless insert is larger than the calculated forces the insert will experience for all operating and accident conditions.

The flexureless insert will perform the same function as the original design and removal of the flexures eliminates the concern for loose flexure heads. Flexureless inserts do not create the potential for any new unanalyzed accidents nor do they increase the probability of any previously analyzed accidents. No changes to the Technical Specifications were required.

The potential for loose parts generation from the flexureless inserts was reviewed by the Staff and found to be of low enough probability to permit their installation in the reactor coolant system.

3.4.128 84-247 (Unit 2), EHC-Megawatt Transducer. This modification installed a flexitest switch in series with the megawatt transducer to allow easy calibration of the megawatt transducer.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.129 84-250 (Unit 2), Reactor Protection System. The modification replaced the existing Sostman loop RTD's with Rosemount Model 176KF and 186-29 RTD's. The Rosemount 176KF RTD's were supplied as "original plant spares" by Westinghouse since Sostman RTD's were no longer available.

Summary of Safety Evaluation: The modification replaces the Unit 2 loop RTD's, which were Sostman, with Rosemount Model 176KF and 186-29, to improve the accuracy of temperature sensors in the reactor coolant pump bypass loops necessitated by the use of optimized fuel assemblies in core reload. The new RTD's are physically adequate, including meeting present temperature and pressure requirements, for replacement. They are not required to meeting environmental qualification and are not contained on the "Master List of Electrical Equipment to be Environmentally Qualified." The Rosemount Model 176KF is fully qualified seismically; and although the Model 186-29 has not undergone seismic testing, it is also considered seismically qualified since this model design which came from the Model 176KF design, did not involve any structural changes to the RTD.

The time response characteristic of the new RTD's is different from the original Sostman RTD's. The Rosemount RTD typically has a time constant of less than 0.6 seconds while the Sostman has a 2-second time constant under similar conditions. Therefore, changes to Tavg and ΔT should be made respective process loops to compensate for the faster time response of the Rosemount RTD's. To accomplish this, Westinghouse has recommended that a 2-second filter be used with the Rosemount RTD per Westinghouse, from an analytical and electrical point of view, there is no difference between use of a Sostman RTD with no filter and a Rosemount Model 176 RTD with a 2-second filter (Rosemount Model 186 is equivalent). (Westinghouse modeling has made an allowance for use of a filter constant of up to 2 seconds.) Since equivalent or a more conservative approach is maintained with the use of up to the 2-second filter with the Rosemount RTD, this will be incorporated in the two affected safety grade signals, overpower ΔT and overtemperature ΔT. Adjustments will be made to the actual measured ΔT signals and the affected Tavg signal used in these two safety grade process loops to ensure proper time response. However, to accurately reflect the filter time constant requirement and thus document acceptable values, a change to Unit 2 Technical Specifications 15.2.3.1.B(4) and 15.2.3.1.b(5) for overtemperature ΔT and overpower ΔT will be submitted to reflect the mathematical equivalent of the entire circuit, including all time constants. Installation of the modification will be procedurally controlled.

3.5 Procedure Changes

The following emergency operating procedures were revised during 1984.

3.5.1 EOP-1B, Operation of Reactor Coolant Gas, Revision 0, 02-24-84. This procedure describes the operation of the reactor coolant gas vent system to vent noncondensible gases from the reactor coolant system.

3.5.2 EOP-4A, Reactor Coolant Leak, Revision 7, 02-24-84. Step 6.4 was amended to include two parameters to be followed before the failure of a reactor coolant pump seal has been verified. They included the maximizing of seal injection and component cooling water to the affected pump and maintaining component cooling water flow less than 260 gpm to avoid auto isolation of thermal barrier.

3.5.3 EOP-4A, Reactor Coolant Leak, Revision 8, 06-20-84. The numbering system for the radiation detectors was changed in Step 4.0 to reflect the installation of the new radiation monitoring system.

3.5.4 EOP-5A, Emergency Shutdown, Revision 11, 02-24-84. It was added to Step 7.1(b) that the Duty Technical Advisor must report to the control room after he/she has been notified. Step 7.6 was amended to include that ECL-5 be performed after the withdrawal of shutdown banks. Appendix "A", Section 3, changed the nomenclature from a system description to a valve description.

3.5.5 EOP-6B, Stuck Rod or Malfunctioning Position Indicator, Revision 8, 02-15-84. Step 7.6.2 was amended to begin with "Further evaluation by Reactor Engineering will be necessary to determine the best way to realign the rod to minimize the effects of core power distribution.

3.5.6 EOP-11B, High Airborne Activity Letdown Gas Stripper Building Ventilation Exhaust, Revision 3, 04-06-84. Step 3.1 had monitor designation changed to reflect the new radiation monitoring system. Step 7 was rewritten to both include new steps and delete one old one. The steps included were the "changeout of 2F11A or B portable air sampler filter" and "start portable air sampler and note the time." One step was deleted from Revision 2 because "2RE-211 & 212 cannot be presently switched to sample the purge exhaust stack." Also, Step 7.1.2 had "call Technical Advisor" deleted and the last step was amended to include specific areas to be surveyed and evaluated "(2F11A or B)".

4.0 NUMBER OF PERSONNEL AND MAN-REM BY WORK GROUP AND JOB FUNCTION

4.1 1984

			JOB FUNCTION					
	NUMBER PERSONNEL 20.1 Rem	TOTAL REM PER WORK GROUP	REACTOR OPERATIONS & SURVEILLANCE	ROUTINE MAINTENANCE	INSPECTION ACTIVITIES	SPECIAL MAINTENANCE	WASTE PROCESSING	REFUELING
1. Company Employees								
Operations	75	92.210	53.560	------	13.940	------	17.830	6.880
Peak Maintenance and Maintenance	101	137.335	------	41.980	5.990	54.615	------	34.750
Chemistry & Health Physics	29	59.660	55.430	------	------	------	3.409	0.830
Reactor Engineering	4	2.710	------	------	1.220	------	------	1.490
Instrument & Control	15	10.200	------	2.090	0.090	6.250	------	1.770
Administration, Engineering, Quality & Regulatory Services	20	7.780	2.900	------	4.560	------	------	0.320
2. Contract Workers and Others*	540	427.260	1.250	------	42.330	352.590	30.300	0.790
TOTALS	784	737.155	113.140	44.070	68.130	413.455	51.530	46.830

*Unit 1 steam generator replacement 1984 exposure = 245.650 Rem

5.0 STEAM GENERATOR TUBE INSERVICE INSPECTION

The following is a synopsis of findings resulting from steam generator tube inspections conducted during 1984.

5.1 Unit 1

The steam generators were replaced during the Refueling 11 outage. Preservice eddy current testing was performed in January, 1984.

In the "A" steam generator, 3,211 tubes underwent full-length inspections. Three tubes had already been plugged with welded shop plugs prior to the preservice inspection.

In the "B" steam generator, 3,214 tubes underwent full-length inspections.

Results of Eddy Current Inspection

	"A" SG		"B" SG	
	Inlet	Outlet	Inlet	Outlet
<20%	0	0	0	0
20-29%	0	0	0	0
30-39%	0	0	0	0
40-49%	0	0	0	0
50-59%	0	0	0	0
60-69%	0	0	1	0
70-79%	0	0	0	0
80-89%	0	0	0	0
90-100%	0	0	0	0

"B" Steam Generator Examination Results

The following tube was plugged as a result of eddy current inspections in the "B" steam generator.

Tube	Defect	Location	Origin
R43C40	60%	25" ATS, HL	OD

ATS - Above tubesheet
HL - Hot leg

5.2 Unit 2

Refueling 10 Inservice Inspection

Eddy current testing of the Unit 2 steam generators was conducted from October 4, 1984, to October 10, 1984, during the Unit 2 Refueling 10 outage.

Extent of inspection in each steam generator was as follows:

Extent of Inspection		No. of Tubes (All from Hot Leg)
Full length		154
Through the "U" Bend		9
To the No. 2 TSP		9
To the No. 1 TSP		
Unsleeved tubes	1,434	
Sleeved tubes	159	
		1,593
To the TTS		60
TOTAL		1,825

"B" Steam Generator

Extent of Inspection		No. of Tubes Hot Leg	No. of Tubes Cold Leg
Full length		69	165
Through the "U" Bend		3	---
To the No. 6 TSP		1	5
To the No. 3 TSP		2	---
To the No. 2 TSP		183	---
To the No. 1 TSP			
Unsleeved tubes	1,447		
Sleeved tubes	113		
		1,560	2
To the TTS		11	---
Totals		1,829	172

TSP - Tube Support Plate
TTS - Top of Tubesheet

The following is a summary of the results of the eddy current inspection showing the numbers of tubes with indications in the ranges listed.

Results of Eddy Current Inspection

	"A" SG		"B" SG	
	Inlet	Outlet	Inlet	Outlet
<20%	12	0	2	43
20-29%	4	0	2	23
30-39%	2	0	0	8
40-49%	4	0	0	0
50-59%	0	0	0	0
60-69%	0	0	0	0
70-79%	0	0	1	0
80-89%	5	0	0	0
90-100%	3	0	0	0

"A" Steam Generator Examination Results

The following tubes were plugged or sleeved as a result of eddy current examinations in "A" steam generator:

Tube	Defect	Location	Origin
R20C06	44%	No. 1 TSP, HL	OD
R10C08	UDI	8.8" ATE, HL	OD
R41C40	88%	5.6" ATE, HL	OD
R42C44	83%	6.9" ATE, HL	OD
R40C52	94%	9.1" ATE, HL	OD
R41C54	96%	4.0"-5.9" ATE, HL	OD
R42C54	<40%	7.6"-10.7" ATE, HL	OD
R41C62	83%	6.4" ATE, HL	OD
R40C65	UDI	4.5"-11.5" ATE, HL	OD
R37C66	<40%	3.4" ATE, HL	OD
R19C75	<40%	7.5"-10.8" ATE	OD
R09C78	91%	13.4" ATE, HL	OD
R01C88	88%	4.4"-8.6" ATE, HL	OD
R07C90	84%	8.6" ATE, HL	OD

HL - Hot leg
ATE - Above Tube End
UDI - Undefinable Indication

"B" Steam Generator Examination Results

Following is the list of tubes mechanically plugged in the "B" steam generator as a result of eddy current examinations:

Tube	Defect	Location	Cause
R06C76	78%	14.5" ATE, HL	Crevice Corrosion
R04C83	NDD*	---	---
R07C83	NDD*	---	---

NDD - No Detectable Indications

*1" tank roll only in hot leg - no hard roll

It was determined during eddy current testing that tubes R07C83 and R04C83 were not hard rolled into the tubesheet in the hot leg, but had only received a 1" tack roll. These tubes were plugged for preventative reasons.

"A" Steam Generator
Hot Leg

Row	Column	Indication	Location	Origin	Plugged
18	5	28%	1TSP	OD	No
20	6	44%	1TSP	OD	Yes
10	8	UDI	8.8" ATE	OD	Yes
23	8	34%	1TSP	OD	No
25	8	30%	1TSP	OD	No
4	16	<20%	TTS	OD	No
4	17	25%	TTS	OD	No
3	18	<20%	TTS	OD	No
4	18	<20%	TTS	OD	No
3	19	<20%	TTS	OD	No
4	19	<20%	TTS	OD	No
41	40	88%	5.6" ATE	OD	Yes
42	44	83%	6.9" ATE	OD	Yes
43	50	28%	1TSP	OD	No
40	52	94%	9.1" ATE	OD	Yes
41	54	96%	4.0"-5.9" ATE	OD	Yes
42	54	<40%	7.6"-10.7" ATE	OD	Yes
41	62	83%	6.4" ATE	OD	Yes
40	65	UDI	4.5"-11.5" ATE	OD	Yes
37	66	<40%	3.4" ATE	OD	Yes
4	73	<20%	0.6" ATS	OD	No
19	75	<40%	7.5"-10.8" ATE	OD	Yes
6	76	<20%	TTS	OD	No
3	77	<20%	TTS	OD	No
4	77	27%	TTS	OD	No
9	78	91%	13.4" ATE	OD	Yes
10	78	<20%	TTS	OD	No
4	79	<20%	TTS	OD	No

Row	Column	Indication	Location	Origin	Plugged
10	79	<20%	TTS	OD	No
9	80	<20%	TTS	OD	No
1	88	88%	4.4"-8.6" ATE	OD	Yes
7	90	84%	8.6" ATE	OD	Yes

ATS - Above Tubesheet
TSP - Tube Support Plate

"B" Steam Generator
Hot Leg

Row	Column	Indication	Location	Origin	Plugged
6	16	<20%	TTS	OD	No
5	18	28%	0.5" ATS	OD	No
6	76	78%	14.5" ATE	OD	Yes
10	76	23%	TTS	OD	No
27	83	<20%	1.4" ATS	OD	No

"B" Steam Generator
Cold Leg

Row	Column	Indication	Location	Origin	Plugged
10	2	27%	1TSP	OD	No
11	2	39%	1TSP	OD	No
12	2	27%	1TSP	OD	No
9	26	25%/21%	0.8" ATS/1.1" ATS	OD	No
10	26	24%/23%	1.1" ATS/0.5" ATS	OD	No
9	28	22%	1.1" ATS	OD	No
6	30	<20%/<20%	1.0" ATS/0.3" ATS	OD	No
7	30	29%	0.8" ATS	OD	No
8	30	<20%/<20%/<20%	0.6" ATS/1.1" ATS/1.9" ATS	OD	No
9	30	<20%/29%	1.8" ATS/0.8" ATS	OD	No
10	30	<20%	1.2" ATS	OD	No
18	32	26%	1.0" ATS	OD	No
7	35	<20%/23%	1.7" ATS/0.7" ATS	OD	No
9	35	<20%	0.6" ATS	OD	No
11	36	29%/<20%	1.4" ATS/1.1" ATS	OD	No
7	37	<20%/<20%	1.8" ATS/0.5" ATS	OD	No
7	38	<20%/<20%	1.9" ATS/0.5" ATS	OD	No
8	38	<20%	2.2" ATS	OD	No
7	39	22%	0.4" ATS	OD	No
5	40	33%	0.7" ATS	OD	No
7	40	23%	0.5" ATS	OD	No
10	40	<20%	0.6" ATS	OD	No
11	40	<20%	0.9" ATS	OD	No
12	40	<20%	0.9" ATS	OD	No

"B" Steam Generator
Cold Leg

Row	Column	Indication	Locations	Origin	Plugged
13	40	<20%	0.9" ATS	OD	No
14	40	<20%	0.6" ATS	OD	No
15	40	<20%	0.9" ATS	OD	No
16	40	<20%/<20%	1.2" ATS/0.8" ATS	OD	No
17	40	<20%	0.6" ATS	OD	No
18	40	<20%	0.5" ATS	OD	No
19	40	<20%	0.5" ATS	OD	No
7	41	26%	0.5" ATS	OD	No
7	43	25%	0.6" ATS	OD	No
7	44	33%	0.6" ATS	OD	No
11	44	31%	1.3" ATS	OD	No
12	44	24%	1.1" ATS	OD	No
9	46	39%	1.6" ATS	OD	No
7	47	22%/39%	3.1" ATS/1.5" ATS	OD	No
6	48	26%/34%	1.4" ATS/0.5" ATS	OD	No
7	48	27%	1.6" ATS	OD	No
6	49	<20%/<20%	1.1" ATS/0.4" ATS	OD	No
9	49	26%/<20%	1.7" ATS/1.1" ATS	OD	No
6	50	<20%	0.4" ATS	OD	No
7	50	<20%/<20%	2.1" ATS/0.5" ATS	OD	No
8	50	<20%	0.4" ATS	OD	No
9	50	<20%/<20%	1.1" ATS/0.3" ATS	OD	No
10	50	<20%	1.1" ATS	OD	No
18	50	<20%/22%	1.3" ATS/0.8" ATS	OD	No
19	50	39%	0.6" ATS	OD	No
20	50	<20%/<20%	1.4" ATS/0.6" ATS	OD	No
21	50	<20%	1.1" ATS	OD	No
22	50	<20%	0.9" ATS	OD	No
23	50	23%/25%	1.1" ATS/0.7" ATS	OD	No
24	50	<20%	0.6" ATS	OD	No
6	51	<20%	0.8" ATS	OD	No
9	51	<20%	0.5" ATS	OD	No
9	52	24%	0.3" ATS	OD	No
23	53	<20%	0.7" ATS	OD	No
23	54	<20%	1.0" ATS	OD	No
7	59	27%	0.5" ATS	OD	No
10	60	<20%/<20%	1.4" ATS/0.8" ATS	OD	No
11	60	<20%/<20%	1.6" ATS/0.8" ATS	OD	No
12	60	<20%	1.1" ATS	OD	No
13	60	<20%	1.1" ATS	OD	No
14	60	<20%	1.2" ATS	OD	No
15	60	<20%	1.4" ATS	OD	No
16	60	<20%	1.0" ATS	OD	No
18	60	<20%	0.8" ATS	OD	No
19	60	<20%	0.7" ATS	OD	No

"B" Steam Generator
Cold Leg

Row	Column	Indication	Location	Origin	Plugged
20	60	<20%	0.7" ATS	OD	No
7	70	<20%	0.8" ATS	OD	No
8	70	22%	0.7" ATS	OD	No
9	70	<20%	0.8" ATS	OD	No
10	70	22%	0.8" ATS	OD	No

6.0 REACTOR COOLANT SYSTEM RELIEF VALVE CHALLENGES

There were no challenges to the Unit 1 or Unit 2 reactor coolant system power-operated relief or safety valves during 1984.

DMB

Wisconsin Electric POWER COMPANY
231 W. MICHIGAN, P.O. BOX 2046, MILWAUKEE, WI 53201

March 1, 1985

Mr. J. G. Keppler, Regional Administrator
Office of Inspection and Enforcement,
Region III
U. S. NUCLEAR REGULATORY COMMISSION
799 Roosevelt Road
Glen Ellyn, Illinois 60137

PRIORITY ROUTING

First		Second	
RA		RC	
DRA		EIC	
DRP	orig+1	SGA	
DRS		ML	
DRSS		OL	
DRNA		OI	
		PAO	

FILE

Dear Mr. Keppler:

DOCKET NOS. 50-266 AND 50-301
ANNUAL RESULTS AND DATA REPORT
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Enclosed are two copies of the Annual Results and Data Report for the Point Beach Nuclear Plant, Units 1 and 2, for the year 1984. This report is submitted in accordance with Technical Specification 15.6.9.1.B and pursuant to the requirements of 10 CFR 50.59(b). The report contains information regarding operational highlights of Units 1 and 2, descriptions of facility changes, tests and experiments, personnel occupational exposures, steam generator inservice inspections, and reactor coolant system relief valve challenges.

Very truly yours,

CW Fay

Vice President-Nuclear Power

C. W. Fay

Enclosure

Copies to NRC Resident Inspector
Director, Office of Inspection and Enforcement (40 copies)

MAR 4 1985

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