WISCONSIN ELECTRIC

POWER COMPANY

POINT BEACH NUCLEAR PLANT

UNIT NOS. 1 AND 2

ANNUAL RESULTS AND

DATA REPORT

1978

U.S. Nuclear Regulatory Commission
Docket Nos. 50-266 and 50-301
Facility Operating License Nos.
DPR-24 and DPR-27

PREFACE

This Annual Results and Data Report for 1978 is submitted in accordance with Point Beach Nuclear Plant Unit Nos. 1 and 2, Technical Specification 15.6.9.1.B (Amendment Nos. 31 and 35 of 12-23-77, respectively) and filed under Docket Nos. 50-266 and 50-301 for Facility Operating License Nos. DPR-24 and DPR-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 ten miles north of Two Rivers, Wisconsin, on the west shore of Lake Michigan.

2.0 HIGHLIGHTS

2.1 Unit 1

For the period 01-01-78 through 12-31-78, which included a 22½ day refueling outage, three steam generator repair outages, and a main transformer replacement outage, Unit 1 operated at an average capacity factor of 87.5% and a net efficiency of 32.6%. The unit and reactor availability were 89.9% and 91.4%, respectively. Unit 1 generated its 24 billionth kilowatt hour (gross) on 02-23-78, its 25 billionth kilowatt hour on 05-16-78, and its 26 billionth kilowatt hour on 08-10-78.

2.2 Unit 2

For the period 01-01-78 through 12-31-78, which included a 28 day refueling outage, Unit 2 operated at an average capacity factor of 89.0% and a net efficiency of 32.9%. The unit and reactor availability were 91.8% and 92.8%, respectively. Unit 2 generated its 19 billionth kilowatt hour (gross) on 02-18-78, its 20 billionth kilowatt hour on 06-09-78, its 21 billionth kilowatt hour on 09-01-78, and its 22 billionth kilowatt hour on 11-22-78.

3.0 FACILITY CHANGES, TESTS AND EXPERIMENTS

3.1 Amendments to Facility Operating Licenses

During the year 1978, there were three license amendments issued by the U. S. Nuclear Regulatory Commission to Facility Operating License DPR-24 for Point Beach Unit 1 and four license amendments issued to Facility Operating License DPR-27 for Point Beach Unit 2. These amendments are summarized as follows.

3.1.1 01-23-78, Amendment 32 to DPR-24, Amendment 36 to DPR-27

These amendments consisted of changes in the Technical Specifications that incorporated the fire protection system into the Limiting Conditions for Operation, surveillance requirements and administrative controls.

3.1.2 02-13-78, Amendment 33 to DPR-24, Amendment 37 to DPR-27

These amendments deleted Section 15.6.12, "Respiratory Protection" from the Technical Specifications.

3.1.3 03-17-78, Amendment 38 to DPR-27

This amendment consisted of changes to the Technical Specifications to allow a one-time waiver of the requirement for monthly functional tests of the turbine stop and governor valves until the start of the fourth refueling outage.

3.1.4 04-18-78, Amendment 34 to DPR-24, Amendment 39 to DPR-27

These amendments removed the storage restrictions related to the spent fuel pool cooling capability since recent design changes have rendered such restrictions unnecessary.

3.2 Facility or Procedure Changes Requiring Nuclear Regulatory Commission Approval

One design change was submitted to the Nuclear Regulatory Commission for approval March 21, 1978, and was subsequently amended on June 14, 1978, and September 29, 1978. Technical Specification Change Request No. 54 contained information on M-511. This modification will require changeout of the existing spent fuel storage racks, this increasing the spent fuel pit storage capacity from 351 assemblies to 1,502 assemblies. The Technical Specification change also included the adding of a redundant hoist to the auxiliary building crane. Nuclear Regulatory Commission review of these changes extended into 1979.

There were no procedures at Point Beach Nuclear Plant in 1978 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 1978 which required Nuclear Regulatory Commission approval.

3.4 Design Changes

3.4.1 Unit 1

a. Electrical 480V Supply (E-127)

This modification request changed the alternate power supply to the white bus Unit 1/Unit 2, 1B41 to 2B41/ 2B41 to 1B41.

Summary of Safety Evaluation: This modification gives real power supply redundancy to the white busses.

b. Power Level Relay Arc Suppressors (E-163)

The modification installed arc suppressors on power level relays 1-32A-TG01 and 1-32B-TG01 since they were burning their contacts due to the relatively slow speed at which they operate. The gradual lightening of contact pressure and gradual opening of the contacts was the cause of the significant arcing across the contact.

Summary of Safety Evaluation: While the system of which this relay is a part performs a function similar to that of the P9 circuitry, it is not part of the QA or safeguards logic system. Not nuclear safety related.

c. Steam Generator Blowdown and Sample Line Isolation Valves (E-167)

The request removed the containment isolation trip signal from blowdown line valves CV-2042 and CV-2045 and from sample line valves CV-2083 and CV-2084 to eliminate containment isolation testing requirements for these valves.

Summary of Safety Evaluation: This modification exercises the second option of a Class 2 (outgoing line) penetration as described in the FFDSAR in that a remotely operated stop valve be utilized in place of an automatic trip valve. The trip function will still respond to a high radiation signal.

d. Gai-tronics System (E-178)

A Gai-tronics speaker was installed in the 8' level fan room of the auxiliary building.

Summary of Safety Evaluation: Not nuclear safety related.

e. <u>Circulating Water System (M-358)</u>

A metal walkway was installed in each condenser waterbox to facilitate safe personnel access to perform condenser tube leak evaluations.

<u>Summary of Safety Evaluation</u>: Not nuclear safety related.

f. <u>Sampling System (M-374)</u>

In-line cation conductivity meters were installed to monitor each waterbox for condenser inleakage.

<u>Summary of Safety Evaluation</u>: Not nuclear safety related.

g. <u>Residual Heat Removal Suction Line Vent Valves (M-385)</u>

The request extended the system pressure boundary to the outside of the biological shield by the addition of 3/8" tubing.

<u>Summary of Safety Evaluation</u>: Not nuclear safety related.

h. <u>RCC Storage Basket Dashpot Assembly (M-445)</u>

A dashpot assembly on the RCC storage basket of the RCC change fixture basket was installed. Each assembly consists of 16 dashpot rodlets with a plate 6.75" square.

<u>Summary of Safety Evaluation</u>: An RCC can accidentally free fall if it is inadvertently released by an operator or a mechanical failure. The addition of a dashpot to absorb energy in such an event should improve the possibility of preventing damage to the rods and support vanes.

i. <u>Residual Heat Removal Safety Valve Setpoint Change (M-455)</u>

The modification lowered the setpoints from 625 to 500 psig by installing new springs and changing the nameplate data to offer more conservative low temperature overpressure protection.

<u>Summary of Safety Evaluation</u>: The modification provides more positive low pressure overpressure protection in the shutdown condition.

j. <u>Steam Generator Upper Modifications (M-457)</u>

The modification removed the downcomer resistance plate between the wrapper and shell to improve the recirculation ratio and replaced the present orifice rings over the swirl vane downcomer barrels with ones of reduced diameter.

Summary of Safety Evaluation: The modification increased flow velocities across the steam generator tube plate, thereby reducing sludge accumulation and the resultant tube wastage possibility.

k. Cavity Drain Line (M-472)

A crud trap was installed on the first elbow of the cavity drain line. Cavity crud which collects in this shielded low point will be more readily disposable.

Summary of Safety Evaluation: Not nuclear safety related.

l. Letdown Gas Stripper (M-480)

An isolation valve was added to the line downstream of the filter to prevent the introduction of air into the letdown gas stripper while changing the filter. An isolation valve was added in the pump recirculating line to the gas stripper to isolate circulating water to the gas stripper so the circulating water pump be isolated for maintenance.

Summary of Safety Evaluation: Not nuclear safety related.

m. Phosphate Addition Na-24 Tracer (M-483)

A Na-24 tracer addition system was installed in the phosphate addition system to provide an injection point for Na-24 during steam generator moisture carryover testing.

Summary of Safety Evaluation: Not nuclear safety related.

n. Turbine Lube Oil Extractor (M-521)

An oil separator was installed in the discharge of the turbine main lube oil vapor extractor to supplement the existing separator.

Summary of Safety Evaluation: Not nuclear safety related.

o. Hydrogen Seal Oil System (IC-132)

A duplex strainer and differential pressure alarm were installed in the air side seal oil system.

Summary of Safety Evaluation: Not nuclear safety related.

p. Turbine Overspeed First Out Indication (IC-168)

The turbine overspeed first out alarm was removed from the EH governor protection and connected to the IOPS circuit trip relay.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.2 Unit 2

a. Electrical 480V Supply (E-128)

This modification request changed the alternate power supply to the white bus Unit 2, 1B41 to 2B41/ 2B41 to 1B41.

Summary of Safety Evaluation

This modification gives real power supply redundancy to the white busses.

b. Diesel Generator Sequencer (E-147)

The modification installed a normally closed contact from the bus A05 and A06 undervoltage auxiliary relays to interrupt the seal-in circuit for the safety injection reset relays. "A" bus undervoltage (loss of off-site power) will then cause a cancellation of the safety injections reset and a reinitiation of the safety injection system if the safety injection actuation logic is still made up; this function then being identical to a manual safety injection.

Summary of Safety Evaluation: This modification assures the automatic restarting and resequencing of safeguards equipment in the event of a loss of AC following initial sequencing of a safety injection incident.

c. Power Level Relay Arc Suppressors (E-164)

The modification installed arc suppressors on power level relays 2-32A-TG01 and 2-32B-TG01 since they were burning their contacts due to the relatively slow speed at which they operate. The gradual lightening of contact pressure and gradual opening of the contacts was the cause of the significant arcing across the contacts.

Summary of Safety Evaluation: While the system of which this relay is a part performs a function similar to that of the P9 circuitry, it is not part of the QA or safeguards logic system. Not nuclear safety related.

d. Steam Generator Blowdown and Sample Line Isolation Valves (E-168)

The request removed the containment isolation trip signal from blowdown line valves CV-2042 and CV-2045 and from sample line valves CV-2083 and CV-2084 to eliminate containment isolation testing requirements for these valves.

Summary of Safety Evaluation: This modification exercises the second option of a Class 2 (outgoing line) penetration as described in the FFDSAR in that a remotely-operated stop valve can be utilized in place of an automatic trip valve. The trip function still responds to a high radiation signal.

e. Main Steam Safety Valves and Piping (E-174)

Freeze protection was removed from the main steam safety valves and piping with the exception of the small drain lines. The valves are normally hot and under pressure and the only time the valves and piping contain water is during steam generator leak tests, and thus, the freeze protection serves no beneficial purpose except for the small drain lines where heat tracing will remain.

Summary of Safety Evaluation: Not nuclear safety related.

f. Circulating Water System Outlet Waterbox (M-361)

A 3" diameter, 6" long pipe was installed in the outlet waterbox. The pipe is used to keep a water seal on the outlet valve without running water through the tubes.

Summary of Safety Evaluation: Not nuclear safety related.

g. Sampling System (M-375)

In-line conductivity meters were installed to monitor each waterbox for condenser inleakage.

Summary of Safety Evaluation: Not nuclear safety related.

h. RCC Storage Basket Dashpot Assembly (M-444)

A dashpot assembly on the RCC storage basket of the RCC change fixture basket was installed. Each assembly consists of 16 dashpot rodlets with a plate 6.75" square.

Summary of Safety Evaluation: An RCC can accidently free fall if it is inadvertently released by an operator or a mechanical failure. The addition of a dashpot to absorb energy in such an event should improve the possibility of preventing damage to the rods and support vanes.

i. Residual Heat Removal Safety Valve Setpoint Change (M-456)

The modification lowered the setpoints from 625 to 500 psig by installing new springs and changing the nameplate data to offer more conservative low temperature overpressure protection.

Summary of Safety Evaluation: The modification provides more positive low pressure overpressure protection in the shutdown condition.

j. Steam Generator Upper Modifications (M-458)

The modification removed the downcomer resistance plate between the wrapper and shell to improve the recirculation ratio and replaced the present orifice rings over the swirl vane downcomer barrels with ones of reduced diameter.

Summary of Safety Evaluation: The modification increased flow velocities across the steam generator tube plate, thereby reducing sludge accumulation and the resultant tube wastage possibility.

k. Cavity Drain Line (M-473)

A crud trap was installed on the first elbow of the cavity drain line. Cavity crud which collects in this shielded low point is now more readily disposed of.

Summary of Safety Evaluation: Not nuclear safety related.

l. Safety Injection System (M-476)

Temperature indicators were installed in the refueling water storage tank to record suction temperatures for safety injection spray pumps during performance of ASME Section XI required testing.

Summary of Safety Evaluation: Not nuclear safety related.

m. Letdown Gas Stripper (M-481)

An isolation valve was added to the line downstream of the filter to prevent the introduction of air into the letdown gas stripper while changing the filter. An isolation valve was also added in the pump recirculating line to the gas stripper in order to isolate a circulating water pump for maintenance.

Summary of Safety Evaluation: Not nuclear safety related.

n. Phosphate Addition Na-24 Tracer (M-484)

A Na-24 tracer addition system was installed in the phosphate addition system to provide an injection point for Na-24 during steam generator moisture carryover testing.

Summary of Safety Evaluation: Not nuclear safety related.

o. Purge Exhaust Ducting (M-512)

The removable spoolpiece was modified by installing clips on the section to eliminate the need for a person to crawl out on the ducting to unbolt the spoolpiece and attach the crane hook.

Summary of Safety Evaluation: Not nuclear safety related; however, the possibility of the section dropping out during an accident or seismic event would not constitute a missile of significant magnitude to damage any other containment apparatus in the vicinity.

p. Turbine Lube Oil Extractor (M-522)

An oil separator was installed in the discharge of the turbine main lube oil vapor extractor to supplement the existing separator.

Summary of Safety Evaluation: Not nuclear safety related.

q. Turbine Overspeed First Out Indication (IC-169)

The turbine overspeed first out indication was removed from EH governor protection and connected to the IOPS circuit trip relay.

Summary of Safety Evaluation: Not nuclear safety related.

3.4.3 Common

a. Reactor Coolant Drain Tank Drain Line (M-314)

The drain line was rerouted to the -19' sump with an isolation valve and test connection added for testing purposes. The modification minimizes contamination on the -5' and -19' levels of the auxiliary building.

Summary of Safety Evaluation: Not nuclear safety related.

b. Rerouting of P9 Discharge to Fuel Transfer Canal Suction (M-319)

The modification permits the draining and filling of the transfer canal from the spent fuel pit, and for the pumping of a holdup tank to the transfer canal. The present four-inch suction line used to pump from the holdup tank or transfer canal to the was utilized to provide a crossconnection between the pit pit's suction to fill the canal from the pit drain.

Summary of Safety Evaluation: The modification reduced the complexity of related valve lineups.

c. Waste Evaporator Vacuum Control Valve (M-331)

The request installed a throttle valve in the distillate inlet line to the eductor and a pressure gauge between the throttle valve and eductor. This will improve operator control of the eductor's operation.

Summary of Safety Evaluation: Not nuclear safety related.

d. Screen Wash System (M-368)

The fish basket hoist was replaced by a motor-operated hoist and trolley.

Summary of Safety Evaluation: Not nuclear safety related.

e. Spent Fuel Pit Purification (M-382)

A new piping run and associated isolation and check valves were installed to enable the use of P33, refueling water circulating pump, as the driving force for purification of spent fuel pit water.

Summary of Safety Evaluation: Not nuclear safety related.

f. Spent Fuel Pit Skimmer (M-389)

This modification relocated the skimmer pump to the waste gas compressor cubicle. The move was necessary because the spent fuel pool cooling modification required this space presently occupied by the pump.

Summary of Safety Evaluation: Not nuclear safety related.

g. Security Related Modification (M-412)

No comment.

h. Security Related Modification (M-413)

No comment.

i. Decay Duct Filter (M-428)

The modification installed a new filtration unit on the condenser air ejector vent line. Installation of the modification fulfilled a commitment to provide filtration of air ejector exhaust gases during periods when significant primary system leakage exists such as during a steam generator tube failure.

Summary of Safety Evaluation: Not nuclear safety related.

j. Security Related Modification (M-429)

No comment.

k. Security Related Modification (M-430)

No comment.

l. New Parking Lot (M-431)

A second parking lot was provided near the entrance to the plant.

Summary of Safety Evaluation: Not nuclear safety related.

m. Security Related Modification (M-436)

No comment.

n. Security Related Modification (M-437)

No comment.

o. Demineralized Water System (M-440)

Piping and valves were installed to provide demineralized water to all levels of the blowdown evaporator cubicle.

Summary of Safety Evaluation: Not nuclear safety related.

p. Service Air System (M-441)

Piping and valves were installed in the service air system to provide air to all three levels of the blowdown evaporator cubicle.

Summary of Safety Evaluation: Not nuclear safety related.

q. Service Building Addition (M-449)

Locker room facilities were expanded and the Ready Stores area was increased.

Summary of Safety Evaluation: Not nuclear safety related.

r. Screen Wash System (M-450)

Flushable strainers were installed in the inlet line to the travelling water screen spray nozzles.

Summary of Safety Evaluation: Not nuclear safety related.

s. Pumphouse Ventilation System (M-469)

The inlet louvers were modified in the circulating water pumphouse ventilation system to provide an improved ventilation system.

Summary of Safety Evaluation: Not nuclear safety related.

t. Turbine Building Ventilation and A/C System (M-471)

The four existing closed loop cooling systems associated with W-13, W-14, W-17 and W-29 were modified to facilitate the addition, recirculation and maintenance of a permanent-type antifreeze solution to prevent freezing of the coil during the winter months.

Summary of Safety Evaluation: Not nuclear safety related.

u. Screen Wash Return Line Extension (M-474)

The request extended the screen wash return line to the downstream side of the fish screen in the Unit 2 circulating water discharge, thus reducing the number of trash fish caught on the screen which must subsequently be removed and hauled to the spoil pile.

Summary of Safety Evaluation: Not nuclear safety related.

v. Clearwell Pump Suction Line (M-485)

The request raised the pump suction line in the clearwell tank. This should reduce the amount of sludge carryover into the water treatment plant.

Summary of Safety Evaluation: Not nuclear safety related.

w. Water Treatment System (M-487)

The water supply piping to the lime mixer was replaced with larger diameter piping.

Summary of Safety Evaluation: Not nuclear safety related.

x. Blowdown Evaporator Steam System (M-499)

The steam trap was modified for operation at lower differential pressures.

Summary of Safety Evaluation: Not nuclear safety related.

y. Nitrogen System Pressure Regulator (M-500)

Pressure regulator 1046 on the nitrogen system was removed to eliminate a long piping run no longer necessary.

Summary of Safety Evaluation: Not nuclear safety related.

z. Potable Water System (M-508)

An emergency shower and eyewash station was installed in the sulfuric acid and the sodium hydroxide tank filling area.

Summary of Safety Evaluation: Not nuclear safety related.

aa. Water Treatment Caustic/Acid Tank (M-518)

The cover for the acid and caustic supply valve pit was modified for easier access.

Summary of Safety Evaluation: Not nuclear safety related.

bb. Heating Boilers Chemical Addition Pot (M-519)

A funnel and vent was added to the chemical addition pot of the heating boilers.

Summary of Safety Evaluation: Not nuclear safety related.

cc. Domestic Hot Water System (M-523)

The hot water lines in the potable water room were rerouted to provide a supply for ion filter regeneration needs.

Summary of Safety Evaluation: Not nuclear safety related.

dd. Instrument Air Test Valve (IC-142)

A Whitey valve and tubing were installed on the PI-3005 sensing line to permit testing of the standby instrument air compressor without depressurizing the entire system.

Summary of Safety Evaluation: Not nuclear safety related.

ee. Blowdown Evaporator Conductivity Cell (IC-143)

The distillate cell range was changed from 0-50 µmhos to 0-500 µmhos.

Summary of Safety Evaluation: Not nuclear safety related.

ff. Potable Water System (IC-146)

An "end of run" alarm was installed for the water softener to minimize salt use and to prevent the system from running out of soft water.

Summary of Safety Evaluation: Not nuclear safety related.

3.5 Procedure Changes

All of the following procedures were reviewed per the requirements of 10 CFR 50.59 and evaluated for their safety implications. They were all approved by the Manager's Supervisory Staff.

3.5.1 ICP 2.1 (Unit 1), Reactor Protection and Safeguards Analog, Channels I through IV (Monthly), Revision 10, 01-20-78

Steps were added to place the presurizer pressure controller in manual during test. (Permanent)

3.5.2 ICP 2.1 (Unit 2), Reactor Protection and Safeguards Analog, Channels I through IV (Monthly), Revision 7, 01-20-78

Steps were added to place the pressurizer pressure controller in manual during test. In addition, the ΔT_{sp1} and ΔT_{sp2} setpoints were changed to reflect Technical Specification changes. (Permanent)

3.5.3 ICP 2.1 (Unit 1), Reactor Protection and Safeguards Analog, Channels I through IV (Monthly), Revision 11, 05-30-78

Step 6.45 was corrected to remove all test equipment and Step 6.149 was corrected to return ten P949 trip switches to normal. (Permanent)

3.5.4 ICP 2.1 (Unit 2), Reactor Protection and Safeguards Analog, Channels I through IV (Monthly), Revision 8, 05-30-78

Step 6.45 was corrected to remove all test equipment and Step 6.149 was corrected to return ten P949 trip switches to normal. (Permanent)

3.5.5 ICP 2.1 (Unit 1), Reactor Protection and Safeguards Analog, Channels I through IV (Monthly), Revision 12, 09-18-78

Steps 6.168 and 6.169 were changed to correct a typographical error (F476 to F467). (Permanent)

3.5.6 ICP 2.1 (Unit 2), Reactor Protection and Safeguards Analog, Channels I through IV (Monthly), Revision 9, 09-18-78

Steps 6.168 and 6.169 were changed to correct a typographical error (F476 and F467). (Permanent)

3.5.7 ICP 2.2 (Unit 1), Reactor Protection and Safeguards Analog, Channels I through IV (Biweekly), Revision 10, 01-20-78

Steps were added to place the pressurizer pressure controller in manual during test. (Permanent)

3.5.8 ICP 2.2 (Unit 2), Reactor Protection and Safeguards Analog, Channels I through IV (Biweekly), Revision 7, 01-20-78

Steps were added to place the pressurizer pressure controller in manual during test. In addition, the ΔT_{sp1} and ΔT_{sp2} setpoints were changed to reflect Technical Specification changes. (Permanent)

3.5.9 ICP 2.2 (Unit 1), Reactor Protection and Safeguards Analog, Channels I through IV (Biweekly), Revision 11, 09-18-78

Steps 6.19 and 6.59 were corrected to read "Tavg ma 10.00 ± 0.2 ma". In Step 6.27 the ideal reading was corrected to be $\leq$42.02, while in Step 6.63, the ideal reading was corrected to be $\leq$28.66 increasing. (Permanent)

3.5.10 ICP 2.2 (Unit 2), Reactor Protection and Safeguards Analog, Channels I through IV (Biweekly), Revision 8, 09-18-78

Steps 6.19 and 6.59 were corrected to read "Tavg ma 10.00 ± 0.2 ma". In Step 6.27 the ideal reading was corrected to be $\leq$42.02, while in Step 6.63, the ideal reading was corrected to be $\leq$28.66 increasing. (Permanent)

3.5.11 ICP 2.5, Periodic Test, Safeguards System Logic, Revision 6, 03-02-78

Step 6.2.6.d was changed to correct a typographical error to the reading for Testpoint 13 as 200 ohms versus the previous 700 ohms. (Permanent)

3.5.12 ICP 2.7, Periodic Test, Nuclear Instrumentation Power Range Channels N41, N42, N43 and N44, Revision 3, 03-14-78

A step was added to return the power mismatch switch to normal. (Permanent)

3.5.13 ICP 2.7, Periodic Test, Nuclear Instrumentation Power Range Channels N41, N42, N43 and N44, Revision 4, 10-24-78

The procedure was changed to use 105% current levels to check channel calibration. (Permanent)

3.5.14 ICP 2.8, Nuclear Instrumentation System Power Range Axial Offset Calibration, Revision 2, 10-25-78

The procedure was changed to reflect setting power range currents at 105% current level to permit "adding" to the power level signal at 100% power with test equipment. (Permanent)

3.5.15 ICP 4.1, Calibration Procedure, Reactor Protection and Safeguards Analog Racks, Revision 2, 05-03-78

The procedure was revised to include Unit 2 calibration data. (Permanent)

3.5.16 ICP 4.3, Calibration Procedure, Pressurizer Level Transmitters, Revision 3, 04-13-78

The procedure was revised to delete use of the pigtail on the top of the bellows and the transmitter calibration data was compensated for the change for Unit 2. (Permanent)

3.5.17 ICP 4.3, Calibration Procedure, Pressurizer Level Transmitters, Revision 4, 10-20-78

The procedure was changed to add Unit 1 compensation test pressures and include opening of the sealed leg bellows vent during calibration. (Permanent)

3.5.18 ICP 4.14, Calibration Procedure, Boric Acid Control System, Revision 2, 04-13-78

The full scale mV for boric acid flow for Unit 2 was added to the data sheet. (Permanent)

3.5.19 ICP 4.24, Calibration Procedure, Nuclear Instrumentation System Source Range, Revision 3, 11-02-78

Typographical errors were corrected and clarifying information was provided to the technicians. A note was added to bypass the channel prior to obtaining voltages, identifying the fuses which must be removed. (Permanent)

3.5.20 ICP 4.25, Nuclear Instrumentation System Intermediate Range Channels, Revision 2, 04-13-78

Unit 1 setpoints were corrected in Step 6.7.3. (Permanent)

3.5.21 ICP 4.25, Nuclear Instrumentation System Intermediate Range Channels, Revision 3, 11-02-78

Steps were rearranged to provide clarification to technicians. In Step 6.4.9 a sentence was added to remove the detector cable and replace fuses in order to calibrate the detector without receipt of a signal. New Step 6.4.18 was added to turn off the picoampere sources, read and record the indication on the amperes neutron level meter (10^{-11} amps). In Step 6.5.5 tolerances were corrected to be I.5V DC. In Step 6.4.18 for N35 and N36 ideal amps and voltages were deleted for 10^{-3} because test equipment does not output 10^{-3} amps. In Step 6.7.3 reset setpoint voltages were added. (Permanent)

3.5.22 ICP 4.26, Nuclear Instrumentation System Power Range Channels, Revision 2, 10-20-78

The tolerances on the P8, P9 and low power range setpoint bistables were changed to ±.04V to account for mid-range meter non-linearity. (Permanent)

3.5.23 ICP 4.26, Nuclear Instrumentation System Power Range Channels, Revision 3, 11-27-78

The procedure was revised so the high overpower trip setpoint could be lowered to 80% on a temporary basis from unit startup to until 75% power physics testing is complete; at which time the setpoint is reset to its normal 107%. (Permanent)

3.5.24 ICP 10.2, Removal of a Safeguards or Protection Sensor from Service, Revision 2, 11-14-78

References and clarification notes were added to more completely delineate instructions to technicians. An appendix was added to detail steps to be followed by operating personnel to place the ΔT trips in the trip mode should the power range instrument fail. (Permanent)

3.5.25 EOP-1A, Large Loss of Reactor Coolant, Revision 15, 03-23-78

Step 3.1.10 was changed to note that the control room supply and cleanup fans are stripped upon a containment isolation signal per a recently completed modification. Step 4.4 was revised to close the following valves prior to resetting safety injection or containment isolation: 1296, 3200C, 3245 and 3213. The revision was made in accordance with a commitment made to the NRC in response to IE Bulletin 78-04 concerning snap lock stem-mounted limit switches which were found not to be environmentally qualified for LOCA conditions. (Permanent)

3.5.26 EOP-2A, Steam Line Break, Revision 8, 03-23-78

Step 3.1.10 changes were the same as for the identical step in EOP-1A. Step 4.6 changes were the same as those made for Step 4.4 of EOP-1A. (Permanent)

3.5.27 EOP-3A, Steam Generator Tube Rupture, Revision 9, 03-23-78

Step 3.1.10 was changed per the note for the same step in EOP-1A. Step 4.2 was changed per Step 4.4 of EOP-1A. (Permanent)

3.5.28 EOP-6A, Stuck Rod or Malfunctioning Rod Position Indication, Revision 5, 02-15-78

Steps 3.2, 4.4.2, 4.5.1, 4.5.4, 4.6.3.b, 4.6.3.h, 4.7.1, 4.8 and 4.9 were revised and/or added to provide procedural steps for adjusting insertion limits. (Permanent)

3.5.29 EOP-8B, Refueling Incident-Containment Evacuation, Revision 3, 12-12-76

Section 1.0 was revised to delete references to the Radiation Protection Manual. Procedures previously contained in that document are now contained in Section 6.0 of the Emergency Plan and in the Health Physics Administrative Control Policies and Procedures Manual. (Permanent)

3.5.30 EOP-10A, Control Room Inaccessibility, Revision 6, 05-15-78

The procedure was completely rewritten to incorporate new fire protection requirements. (Permanent)

3.5.31 EOP-11B, High Airborne Activity, Letdown Gas Stripper Building Ventilation Exhaust, Revision 1, 01-20-78

In the original issue of this procedure a step had been included regarding the R11/R12 monitor; however, this step had not been completely described. This revision required that R11/R12 sampling must be switched from the containment to the stack which will monitor the release. In addition, several steps were rearranged to better perform the procedure. (Permanent)

3.5.32 OP-1A, Cold Shutdown to Low Power Operation, Revision 15 04-28-77 (performed on Unit 2 04-15-78

In Step 4.13.7 primary system pressure was changed from 2335 psig to 2370 psig to facilitate a hydrostatic test of a safety injection check valve and comments were added to clarify temperature and pressure requirements. Steps 4.18, 4.19, 4.21, 4.29 and 4.30 were not applicable and were modified in accordance with the provisions of WMTP2 3.2 which had already received Staff approval. (Temporary)

3.5.33 OP-1A, Cold Shutdown to Low Power Operation, Revision 16, 07-12-78

Step 2.25 was added to include the maximum heatup rate of the pressurizer as listed in the Technical Specifications. Step 4.3.1 was added for safety concerning reactor coolant pump motor flywheel covers. (Permanent)

3.5.34 OP-1A, Cold Shutdown to Low Power Operation, Revision 17, 10-09-78

Step 4.13 was changed to call out performance of IT-230 if required (for leak testing after refueling per Section XI). Step 4.34 was added to inspect the decon valves (543 and 544) for packing leakage. A caution note from the reactor coolant pump technical manual was added to Step 4.3 that the seal bypass valve should not be opened unless the seal return valves are open; this preventing lifting of the seal ring off the runner which can cause the seal ring to hang up. (Permanent)

3.5.35 OP-1A, Cold Shutdown to Low Power Operation, Revision 17, 10-09-78 (performed 10-11-78)

Step 4.25.1 was added to return the safety injection pumps to service; the philosophy being to help prevent overpressurization due to inadvertent safety injection pump activation. Thus, the pumps were returned to service after system pressure was returned and a bubble had been established in the pressurizer. Step 4.13.8 was accomplished during performance of IT-230 and thus was not required during performance of this procedure. (Temporary)

3.5.36 OP-1A, Cold Shutdown to Low Power Operation, Revision 18, 12-26-78

Step 2.5 changed the reference to shutdown margin to the amount of shutdown as required. A reference was added in Step 2.17 to assure proper pressurizer chemistry. In Step 2.32.2 and 4.5 the reference to part-length rods was deleted since the rods have been removed. Steps 3.1.8 and 3.1.9 were corrected to require portions of related checklists. Steps 3.5 through 3.7 were revised to ensure proper lineup of the steam system. Steps 3.8 through 3.10 were added to ensure proper ventilation is started. In Step 4.3.1 the reference to a period of hot shutdown to that of operation after which CV-386, No. 1 seal bypass, would be shut, was corrected. In Step 4.6 a qualifying phrase was added to shut previously opened valves. The remainder of the procedure was rewritten to reorganize and streamline numerous previously made changes due to ASME Section XI Code requirements and controls for the prevention of an overpressurization incident. (Permanent)

3.5.37 OP-1C, Low Power to Normal Power Operation, Revision 14, 12-21-76 (performed 04-18-78)

In Step 3.6, "procedure OP-14B" was changed to "checklist CL-14B" since the operating procedure was canceled. The checklist remains valid. (Temporary)

3.5.38 OP-1C, Low Power to Normal Power Operation, Revision 15, 04-19-78

The temporary change noted above was made permanent in the next revision. Step 2.34 was revised to maintain lube oil cooler outlet temperature at 80^{0}F per Westinghouse recommendations. Step 3.11 was revised to add a comment on maintaining lube oil temperature at 80^{0}F per Westinghouse recommendation. New Step 4.1.3 was also added to ensure main lube oil cooler TCV is in auto and set for warming up lube oil to normal operating temperature. (Permanent)

3.5.39 OP-3A, Normal Power Operation to Low Power Operation, Revision 5, 10-21-74

Steps 4.1, 4.2 and 4.3 were not performed when this procedure was used following return of Unit 2 to the line on January 10, 1978, after the intake freezeup. The unit had not been up in power long enough to have REI 1.0 performed (primary system heat balance), thus leaving the nuclear instrumentation system valid for the low power operation. (Temporary)

3.5.40 OP-3A, Normal Power Operation to Low Power Operation, Revision 5, 10-21-74 (performed 10-15-78)

Steps 4.18 through 4.23 were deleted because turbine overspeed testing was being performed in conjunction with returning Unit 1 to service following its refueling outage. (Temporary)

3.5.41 OP-3B, Reactor Shutdown, Revision 5, 05-12-78

A typographical error was corrected in Step 2.7. Step 4.6 which advises the operator to borate per OP-5B was deleted and subsequent steps renumbered since the information is contained in OP-5B. (Permanent)

3.5.42 OP-3C, Hot Shutdown to Cold Shutdown, Revision 16, 11-02-77 (changes made for Unit 1 "A" steam generator repair outage on 02-02-78)

Steps 4.5.1 and 4.5.2 which perform inservice testing on the auxiliary feedwater valves (IT-140) and pumps IT-8 and IT-10), respectively, were delayed until unit startup. (Temporary)

3.5.43 OP-3C, Hot Shutdown to Cold Shutdown, Revision 17, 03-10-78

Typographical errors were corrected in Steps 2.5 and 2.9. Step 3.4 was added to reach hydrogen concentrations of 25 to 30 cc/kg prior to cooldown. A sentence was added to Step 4.9 to perform IT-200 (Unit 1) or IT-205 (Unit 2), Test of Safety Injection Accumulator Discharge Check Valves. A note was added to Step 4.12.1 to indicate <10 cc/kg hydrogen in the reactor coolant system prior to opening the system. (Permanent)

3.5.44 OP-3C, Hot Shutdown to Cold Shutdown, Revision 17, 03-17-78 (performed 03-23-78)

Step 3.5 was added to assure rod drops were done prior to cooling down. Step 4.13.a was added to call out ORT #8 to be performed, visual check of residual heat removal system following placing residual heat removal into service. Steps 4.14.a, 4.15.a and 4.15.b were added to concur with chemistry requirements. Step 4.17.a was added to permit the containment integrated leak rate test. Step 4.20 corrected a typographical error (IT-115 versus IT-144). Step 3.2 was deleted because shutdown banks were withdrawn as allowed in Step 2.11. (Temporary)

3.5.45 OP-3C, Hot Shutdown to Cold Shutdown, Revision 18, 05-16-78

Section 1.0 was revised to delete all references to refueling shutdown since the procedure does not apply to these. Step 2.8 was revised to permit boration through normal makeup that is required during cooldown due to shrink of the system. Step 2.9 was changed to provide guidelines to minimize reactor coolant system crud buildup. Step 2.10 was removed because the Staff felt the original Westinghouse recommendation need not be proceduralized as it is precluded by design and normal operating practice. Step 3.2 was deleted since boration occurs during cooldown. In Step 4.2 the reference to boron concentration was removed. New Step 4.3.1 was added to call up testing of the non-return valves per IT-220 or IT-225. In Step 4.10 a note was added to call out valving in the low pressure transmitters for low pressure protection. In Step 4.15 the words "commence to" were deleted for clarification purposes. In Step 4.17 the note was changed such that a typographical error was corrected and the individual step (4.14) in procedure OP-7A was deleted to avoid confusion should that procedure be revised and renumbered. In Step 4.20 a typographical error for an IT test was corrected. (Permanent)

3.5.46 OP-3C, Hot Shutdown to Cold Shutdown, Revision 18, 05-16-78 (procedure conducted on 05-26-78, Unit 1)

In Step 4.12.1, the allowable hydrogen concentration was changed from less than 10 cc/kg to 11.6 cc/kg for securing of the letdown gas stripper. The change was necessitated because of the erratic hydrogen readings being obtained during the shutdown. It was determined that by proceeding with the shutdown there would be no nuclear safety related problems. (Temporary)

3.5.47 OP-3C, Hot Shutdown to Cold Shutdown, Revision 19, 07-20-78

Steps 2.8, 2.9 and 4.1 were changed to minimize transport of crud from the core to the steam generators and reduce radiation levels in the generators. (Permanent)

3.5.48 OP-3C, Hot Shutdown to Cold Shutdown, Revision 20, 09-29-78

References to IT-8, IT-9 amd IT-10 were deleted for cold shutdown conditions while IT-140 and IT-145 were added for the inservice testing of the auxiliary feedwater system. A note was added to Step 4.21 allowing the deletion of testing of valves 854A and B (refueling water storage tank residual heat removal suction checks) since the inservice tests (IT-120 and IT-125) will be performed during ORT #2 if the unit is in a refueling shutdown. (Permanent)

3.5.49 OP-4A, Filling and Venting Reactor Coolant System, Revision 11, 04-28-77

Changes for Unit 1 recovery on 02-05-78 following tube plugging in the "A" steam generator. Step 4.18, requiring venting of all reactor coolant pump instrumentation was deleted because it was not necessary. (Temporary)

3.5.50 OP-4A, Filling and Draining the Reactor Coolant System, Revision 11, 04-28-77 (procedure conducted 05-29-78, Unit 1)

In Step 4.12 a typographical error was corrected such that valve RC-525 was changed to 522A. (Temporary)

3.5.51 OP-4A, Fill and Vent of the Reactor Coolant System, Revision 11, 04-28-77 (to be performed 10-10-78)

Step 4.17.1 was added to flush the drain valve lines to the pressurizer relief tank in preparation for performance of IT-230. (Temporary)

3.5.52 OP-4A, Filling and Venting Reactor Coolant System, Revision 12, 09-21-78

A new Step 4.10 was added to prevent the unnecessary processing of water drained from the reactor coolant system, especially for a non-refueling shutdown. Old Step 4.10 was deleted because it is covered by CL-4A in Step 3.1. In Step 4.12 a typographical error was corrected. In Step 4.13 the valves called out in Step 3.1 of CL-4A were noted to be placed in the shut position; these being the valves at the pressurizer relief tank. In Step 4.17, the loop valves are closed, an operation which can be done in conjunction with venting of the reactor coolant pumps during a loop entry. Step 4.30 was added to shut IVTC-1421, pressurizer relief tank to the purge exhaust. This step had been inadvertently omitted from previous issues of the procedure. (Permanent)

3.5.53 OP-4D, Draining the Reactor Coolant System, Revision 12, 09-20-78

Step 4.8 was revised to correct valve numbers to coincide with the valves cited. Steps 4.15, 4.20 and 5.25 were added to reflect changes made in the venting system. Step 4.16 was reworded to clarify the use of the new venting system. Section 5.0 was rewritten to allow draining of the refueling cavity to the refueling water storage tank via the residual heat removal system, allowing the pressurizer to be vented while draining. An addition was made to Step 6.1 to assure the suction hose manway adapter is attached to the suction hose. A reference was added to Step 6.2 to assure proper hookup of ducting equipment to the hot leg. Step 6.3 was added to ensure proper fan operation. (Permanent)

3.5.54 OP-4D, Draining the Reactor Coolant System, Revision 12, 09-20-78 (temporary change made 09-29-78)

In Step 3.12 since three charcoal trays were not in place, they could not be made operable. (Temporary)

3.5.55 OP-5A, Reactor Coolant Volume Control, Revision 7, 04-28-77 (performed 10-12-78)

Although Revision 8 to the procedure had been approved by the Staff, copies of the new revision had not yet been distributed for use. Therefore, all permanent changes made for that revision were made to Revision 7 on a temporary basis for use by operating personnel. The changes were made per a Westinghouse technical bulletin to aid in preventing lifting of the seal ring off the runner. (Temporary)

3.5.56 OP-5A, Reactor Coolant Volume Control, Revision 8, 10-09-78

A caution note was added to Step 4.6 to include reactor coolant pump technical manual supplementary information as noted above for Step 4.3 of OP-1A. In addition, the sequence of Step 4.6 and 4.5 was reversed based upon operational experience. (Permanent)

3.5.57 OP-5D, Chemical Addition and Control, Revision 5, 05-12-78

In A2.3, B2.3, and C2.2, reactor coolant pump operating conditions were revised to be as per OP-4B rather than seal return pressure being greater than 15 psig when reactor coolant pumps are in operation. This was done to preclude the need of having to revise this procedure in the future when changes are made to OP-4B. (Permanent)

3.5.58 OP-7A, Placing Residual Heat Removal System in Operation, Revision 10, 05-12-78

In Steps 4.1.3 and 4.1.4, typographical errors were corrected. New Steps 4.5 through 4.8 were added and subsequent steps renumbered to minimize level transients in the pressurizer while placing residual heat removal on the line. (Permanent)

3.5.59 OP-7A, Placing Residual Heat Removal System in Operation, Revision 11, 09-07-78

Step 4.16 was added to the procedure to remind personnel to perform IT-120 (Unit 1) or IT-125 (Unit 2), Step 4.50 for stroke verification of check valve 867B. (Permanent)

3.5.60 OP-7A, Placing Residual Heat Removal System in Operation, Revision 12, 09-27-78

Notes were inserted following Steps 4.8 and 4.14 to call out the requirement to perform ORT #8, residual heat removal leak check during the refueling shutdown. (Permanent)

3.5.61 OP-7B, Removing Residual Heat Removal System from Operation, Revision 6, 09-07-78

Step 4.10 through 4.11 were modified to record inservice test data as a means of verifying flow through check valves 854A&B. (Permanent)

3.5.62 OP-8A, Spent Fuel Pit Cooling Water System Operation, Revision 4, 05-15-78

Steps 2.1, 2.2, 2.3, 3.3, 5.1.2, 4.2.2 were revised to accommodate the new spent fuel pit cooling system as installed per modification request M-278. (Permanent)

3.5.63 OP-9B, Solid Waste Processing, Revision 10, 07-25-78

Steps 2.6, 2.9, 4.1.5, 4.3.8, 4.5.1 and 4.5.2 were changed to correct typographical errors and to add clarification to the procedure. Additional changes were made to provide a more reliable waste records system. (Permanent)

3.5.64 OP-9C, Containment Venting and Purging, Revision 7, 12-07-77

Steps 2.2.3, 2.2.6, 4.2.1-4.2.10 were revised and/or rearranged to clarify implementation of the procedure and several terminology changes were made to better accommodate actual nomenclature used at the plant. (Permanent)

3.5.65 OP-9D, Isolation, Decay and Discharge of Gaseous Wastes, Revision 3, 07-25-78

Section 3.0 and 4.0 were rewritten to clarify valve lineups for releases and Attachment "A" was added to provide a standard isolation tag series. (Permanent)

3.5.66 OP-13A, Secondary Systems Startup and Shutdown, Revision 11, 01-23-78

Steps 2.8 and 2.9 were changed and the remaining steps renumbered. Step 2.8 notes that whenever the second circulating pump is shut down on a unit, steam will be taken out of the affected turbine unless the pump can be restored within five to ten minutes. This change was made based upon experience gained during recent circulating water system problems, including unit freezeup. Step 2.9 notes that valve CV-28, the manual bypass around the air ejector and gland steam condensers which is usually shut, may be used to increase steam generator feed pump suction if required. The change was made in view of the presently occurring condensate pump boot leakage problems. (Permanent)

3.5.67 OP-13A, Secondary Systems Startup and Shutdown, Revision 12, 06-30-78

New Steps 2.11, a note after 4.10.2, Step 4.10.5 and 5.9.5 were added, with Step 5.9.5 being modified in order to accommodate the new combined air ejector delay duct filter unit and to preclude possible condenser vacuum problems. (Permanent)

3.5.68 OP-13A, Secondary Systems Startup and Shutdown, Revision 12, 06-30-78 (performed 09-20-78)

Step 5.2 which tests the atmospheric steam dumps was waived to keep radiation releases to a minimum following the steam generator tube leak. (Temporary)

3.5.69 OP-13A, Secondary Systems Startup and Shutdown, Revision 13, 08-11-78

In Step 2.6 a comment was added to maintain lube oil temperature at 30°F per Westinghouse recommendation. New Steps 4.5.4 and 5.15 were added to include the new condenser hotwell sampling system installation.

3.5.70 OP-13A, Secondary Systems Startup and Shutdown, Revision 12, 06-30-78 (performed 10-14-78)

In Step 3.1.6 a typographical error was corrected to indicate CL-10C. In Step 4.6, reference to ORT #4 was replaced with IT-150 and IT-155 for testing of the main steam stop valves. Revision 12 of the procedure was used since Revision 13 had not been distributed. The three changes which were made in Revision 13 were minor in nature and were not nuclear safety related. The changes called for the startup and shutdown of the new hotwell sampling system and maintaining lube oil temperature at 80°F per Westinghouse recommendations; these recommendations having been standard operating practice. (Temporary)

3.5.71 OP-13A, Secondary Systems Startup and Shutdown, Revision 14, 10-24-78

The temporary changes listed for Steps 3.1.6 and 4.6 were made permanent in this formal revision to the procedure. (Permanent)

3.5.72 OP-14B, Turbine Lube Oil System Startup, Revision 2, 01-15-76

This procedure had been used for initial startup of the lube oil systems and that since early in operation, a periodic check has been used in its place. For the procedure to be an OP (major) is questionable since it involves the use of a non-nuclear safety related system. Therefore, the procedure was canceled. (Permanent)

3.5.73 OP-14C, Lube Oil Transfer Procedure, Revision 1, 06-02-75

This procedure, also non-nuclear safety related, would be more appropriate in the form of an operating instruction rather than a formal major OP. This procedure was canceled and replaced with OI-48. (Permanent)

3.5.74 RP-1A, Preparation for Refueling, Revision 9, 12-01-77

Step 4.5 deleted the requirement for disconnecting part-length rods since they were removed during the last refueling. In Step 4.30.a a note was added to indicate inservice inspection of the reactor vessel flange. (Permanent)

3.5.75 RP-1C, Refueling, Revision 5, 12-21-76

In Step 4.13, the following computer addresses were changed for Unit 2 trending: residual heat removal flow F0626A to F0626C; source range detector 21 N0031A to N0031C; source range detector 32 N0032A to N0032C; and since spent fuel pit radiation monitor R0005A is not on the Unit 2 computer and could not be trended, it was deleted. Four additional steps were added to verify the complete freedom of the primary source assemblies since another facility had two of its primary sources stick in fuel assemblies. These included Step 312 and 313 which transferred E-08 from G-12 to center to G-12; and 314 and 315 which transferred E-17 from G-2 to center to G-2. (Temporary)

3.5.76 RP-1C, Refueling, Revision 6, 07-20-78

Step 1.5 was changed from OI-8 to ORT-15. Step 4.12 was added to prevent access to the transfer tube during the shuffling of fuel. Step 5.3.2 removed consideration for statistical nature of counts as this is taken care of in the scale of the plot and interpretation of changes. In Step 6.9 the statement referencing the use of a computer form was removed. (Permanent)

3.5.77 RP-1D, Filling and Draining the Refueling Cavity, Revision 4, 04-19-78

Valves were added to Step 6.2 in accordance with modification request M-278 for the new spent fuel pit cooling system.

3.5.78 RP-2C, New Fuel Inspection, Revision 3, 09-18-78

Throughout the procedure, all references to TV inspection were removed since the TV is no longer used. All references to use of the scale fixture were also deleted for this same reason. All references to the polyethylene wrapper were also deleted because assemblies are no longer wrapped. In Step 4.8.2 a sentence was added to check the fuel assembly face for straightness. Step 6.10 was deleted because the Reactor Engineering Group is the sole responsible party for fuel inspection and the step thus was not needed. (Permanent)

3.5.79 RP-4A, Full-Length Control Rod Drive Shaft Unlatching and Latching, Revision 1, 04-07-78

In its first revision since original issue some four years ago, a general rewrite was performed for clarification purposes. There was no change in the intent of the procedure. (Permanent)

3.5.80 HP 1.5, Exposure Limits for Non-Plant Personnel, Revision 2, 04-26-78

Section 1.0 was changed to define contractor and visitors as normally monitored "exempt" from 10 CFR 20.202. (Permanent)

3.5.81 HP 2.1, Designation of and Access to the Controlled Zone, Revision 1, 05-17-78

Section 1.0 was changed to include "exempt category". Section 3.0 was added to define Checkpoint "A" and "B" with regards to female access to the controlled side of the plant. (Permanent)

3.5.82 HP 7.1, Federal Agencies and Insurance Personnel, Revision 2, 04-11-78

The procedure's title, Section 1.0 and 2.0 were revised to accommodate notification of Company insurance personnel in addition to notification of governmental agencies in the event of a radiation incident or emergency. (Permanent)

3.5.83 HP 11.4, Radioactive Source Inventory and Sealed Source Leak Testing Procedure, Revision 2, 11-20-78

The procedure was revised to include the inventory of radioactive sources. The designation of the procedure was changed from minor to major. (Permanent)

3.5.84 HP 13.1, Monthly Operational Test of the Radiation Monitoring System (Area and Process Subsystems), Revision 1, 04-20-78

Steps 5.2, 5.4, 5.5, 5.6, 5.7, 5.8 and 5.10 were revised to provide an electronics check and to use, when necessary, a check source from health physics supplies when the installed check source is inoperative or too weak to cause a reading. (Permanent)

3.5.85 HP 13.1, Monthly Operational Test of the Radiation Monitoring System (Area and Process Subsystems), Revision 1, 04-20-78

References were added to conduct the test following completion of refueling procedure RP-1C. In Step 5.1 certain process monitors were removed from scan on the health physics computer program before testing so a spike would not occur when the monitors were simply being calibrated. (Permanent)

3.5.86 Emergency Plan Manual, Section 6, Major Radiation Emergencies, Revision 8, 09-15-78

Changes to Section 6 of the Emergency Plan which are defined as required procedures by the Point Beach Nuclear Plant Technical Specifications are as follows: (1) correction of minor editorial and typographical errors, (2) Changes regarding the incorporation of ERDA into DOE, and (3) changes reflecting the merger of Wisconsin Michigan Power Company with Wisconsin Electric Power Company. Certain position titles were also changed as a result of this merger.

A Revision 9, dated 12-08-78, involved no changes to Section 6; and, therefore, are not reportable.

3.5.87 WMTP1 2.1, Core Refueling, Revision 0, 08-25-78

D-85 was moved from S-26 to R-6 in the spent fuel pool to facilitate changing the burnable poison. This had been omitted in the prior to core loading preparations. Instead of adding a fuel movement from the upender to the end basket to change the plug device, the following steps were changed to change the plug device to the upender instead. The change eliminated one fuel handling step. 134, 135B, 136, 140, 141B, 142, 146, 147B, 148, 154, 156B, 157, 164, 165B, 166, 172, 173B, 174, 180, 181B, 182, 189, 190B, 191, 196, 197B, 198, 204, 205B, 206, 210, 211B, 212, 259A, 259B, 260. There were seven additional steps done because of the H-10 incident (assembly bumped). The two fuel assemblies (H-10 and G-03) were taken to spent fuel pit locations R-6 and R-7, respectively, inspected and returned. Step 4.14 of RP-1C relates to the source range calibration. Only Channel 31 was done prior to the start of refueling. Channel 32 became available at Step 14 (1511 hours, 10-03-78). (Temporary)

3.5.88 WMTP1 2.5, Irradiation Specimen Shipment, Revision 0, 01-05-78

Step 5.1.1 no block was removed from the truck because it was not on it. Steps 5.1.2, 5.1.3 and 5.1.4 the cask was upended from the cradle. The cradle had a lip at the end where the base of the cask could pivot on when upending. Step 5.1.8 no choker was attached because a wire was attached to the lifting beam. A shackle was used to attach the eye at the end of the wire to the lead plug. Step 5.1.9 the lead plug was placed on the decon pad grating. Step 5.1.10 there was no drain valve provided. The drain was open. Step 5.2.1 space was left to take the lifting beam off to the north. Step 5.4.4 a three-eighth inch Swagelok cap was used to seal the drain. Steps 5.4.8 and 5.4.9 the cask was directly laid into the cradle. (Temporary)

3.5.89 WMTP1 3.2, Primary System Tests - Hot, Refueling 6, Revision 0, 08-25-78

The procedure calls out WMTP 9.1. In Step 5.3.2 even though the voltage signals at 20 steps did not fall within the review criteria, the zero and span were not adjusted. The position indicators appeared to respond well without adjustment. (Temporary)

3.5.90 WMTP 9.1, Rod Control Mechanism Timing, Rod Drop and Rod Position Calibration, Revision 5, 01-10-77

The procedure was classified as a major procedure but in the Refueling 6 startup test application, it was used as a supporting document to WMTP1 3.2 and therefore, the procedure was reclassified during the Refueling 6 startup test package. (Temporary)

3.5.91 WMTP1 4.3, Rod Worth Measurements by Swap Method, Refueling 6, Revision 0, 10-10-78

The date on the procedure contained a typographical error and is correctly 10-10-78. Steps 5.6, 5.7 and 5.8 were not performed because of (1) time considerations, and (2) they would have been a second measurement of rod worth which was not required. (Temporary)

3.5.92 WMTP1 5.3, Power Level Increase to Full Load, Revision 0, 08-25-78

In Step 5.9 power was reduced for 12 hours to 90% instead of 24 hours because it was not necessary to hold the unit for a longer period to achieve equilibrium xenon and because of Power Supply consideration. (Temporary)

3.5.93 WMTP2 2.1, Core Reloading, Revision 0, 03-03-78

Four additional steps were added to verify the complete freedom of the primary source assemblies. (Temporary)

Step	Assembly	From	To
312	E-08	G-12	Center
313	E-08	Center	G-12
314	E-17	G-02	Center
315	E-17	Center	G-02

3.5.94 WMTP2 6.1, Unit 2 End of Cycle 4 Physics Tests, Revision 0, 03-14-78

Step 5.1 which measures voltage and current to each of the pressurizer heater groups was not performed because of breaker problems. The measurements will be taken upon startup. (Temporary)

3.5.95 WMTP 9.1, Rod Control Mechanism Timing Rod Drop and Rod Position Indication, Revision 5, 01-10-77

Initial condition 4.6 was deleted during the Unit 2 end-of-cycle 4 physics tests. Baseline counts were not taken because the reactor was critical. (Temporary)

3.5.96 WMTP 9.8, Xenon Follow Measurement, Revision 3, 08-17-76

Step 4.2 was deleted during the Unit 2 end-of-cycle 4 physics tests. The boron concentration in the computer was not updated because no boration was planned. (Temporary)

3.5.97 WMTP 9.2, Power Range Calibration Quarterly Axial Offset Test, Revision 7, 09-13-78

Steps 3.7 and 3.8 were added to specify that ΔØ maneuvering instructions should be entered in the night orders when ΔØ maneuvering is done on backshifts, the unit should remain at full power between flux maps and that the precautions and limitations of REI 6.0 apply when a flux map is to be taken. Step 5.6 was added instructing to use REI 6.0 to obtain quarter core flux maps during ΔØ swings at the option of the Test Engineer. Step 5.9 was revised to consider taking quarter core flux maps. (Permanent)

3.5.98 WMTP 9.4, Initial Criticality For a Cycle, Revision 4, 01-20-78

Step 5.8 had two more entries made available for different dilution rates. Only one entry was previously available. Step 5.11 was added since it may be desirable to slow or stop the dilution rate to allow pressurizer boron concentration to mix with the loops. Step 5.16.4 was changed from "60 pcm and 80 pcm" to "about 60 pcm and 80 pcm". Step 5.17.1 was changed to reference RE 5.0 or the design report for rod worth data. Step 5.19 was added defining a method to determine the effect of flux level on reactivity. (Permanent)

3.5.99 WMTP 9.7, Power Coefficient Measurement, Revision 4, 02-09-78

Steps 4.9-5.7 were reworked to streamline the procedure and clarify certain steps.

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

4.1 1977 (Corrected)

WORK GROUP	NUMBER PERSONNEL >100 mRem	TOTAL REM PER WORK GROUP	JOB FUNCTION: REACTOR OPERATIONS & SURVEILLANCE	ROUTINE MAINTENANCE	INSPECTION ACTIVITIES	SPECIAL MAINTENANCE	WASTE PROCESSING	REFUELING
1. Company Employees								
Operations	43*	78.355[1]	38.336	1.763	14.075	2.873	9.263	12.045
Peak Maintenance and Maintenance	63*[1]	119.820[1]	0.0	59.517	26.005	34.298	0.0	0.0
Chemistry and Health Physics	18*	33.380[1]	28.606	1.706	0.572	0.182	1.645	0.669
Reactor Engineering	5	3.556[1]	1.306	0.0	0.736	0.166	0.0	1.348
Instrument and Control	12[1]	2.266	0.215	1.795	0.256	0.0	0.0	0.0
Administration	5	0.956	0.199	0.0	0.757	0.0	0.0	0.0
2. Contract Workers and Others	203	232.379	0.0	5.856	7.296	219.227	0.0	0.0
TOTALS	305	470.712[1]	68.662	70.637	49.697[1]	256.746[1]	10.908	14.062

* Intercompany transfers people counted only once, not as one person in each group.

[1] Corrected information as reported in the 1977 Annual Report.

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

4.2 1978

			JOB FUNCTION					
	NUMBER PERSONNEL ≥100 mRem	TOTAL REM PER WORK GROUP	REACTOR OPERATIONS & SURVEILLANCE	ROUTINE MAINTENANCE	INSPECTION ACTIVITIES	SPECIAL MAINTENANCE	WASTE PROCESSING	REFUELING
1. Company Employees								
Operations	43*	77.196	44.651	0.141	8.428	0.895	17.942	5.139
Peak Maintenance and Maintenance	76	92.312	0.488	37.579	27.718	26.054	0.326	0.147
Chemistry and Health Physics	24*	23.969	21.418	0.318	0.023	0.0	1.677	0.533
Reactor Engineering	3	3.524	1.920	0.0	1.285	0.0	0.0	0.319
Instrument and Control	7*	3.638	0.0	2.391	1.053	0.194	0.0	0.0
Administration	3	1.103	0.0	0.0	1.103	0.0	0.0	0.0
2. Contract Workers and Others	128	107.026	0.0	77.496	0.0	29.530	0.0	0.0
TOTALS	284	308.768	68.477	117.925	39.610	56.673	19.945	6.138

* Intercompany transfers people counted only once, not as one person in each group.

5.0 STEAM GENERATOR TUBE INSERVICE INSPECTION

5.1 Unit 1

02-02-78 Outage

On 02-02-78 the unit was shut down with a primary-to-secondary leak rate of 130 gpd. The source of the leak was verified by a leak test to be a tube at position R43C32 in the "A" steam generator. The tube was explosively plugged. Following verification of the integrity of the plug by a second leak test, the unit was returned to the line on 02-06-78. No eddy current examination was conducted during the outage.

05-25-78 Outage

On 05-25-78 the unit was shut down with an estimated primary-to-secondary leak rate of 145 gpd. The shut down was made in conjunction with a previously scheduled Memorial Day weekend maintenance outage. A secondary-to-primary leak test was conducted which diclosed a leaking tube in the "A" steam generator at position R44C35. The tube was explosively plugged. Following a leak test which verified the integrity of the repair, the unit returned to the line on 05-30-78. No eddy current examination was conducted during the outage.

Refueling 6 Inservice Inspection

Extent of Eddy Current Inspection

"A" SG	Inlet	214 tubes tested at 400 KHZ through U-bend
		780 tubes tested at 400 KHZ through first support
		689 tubes tested at 400 KHZ through sixth support
	Outlet	15 tubes tested at 400 KHZ through U-bend
		0 tubes tested at 400 KHZ through first support
		154 tubes tested at 400 KHZ through sixth support
"B" SG	Inlet	119 tubes tested at 400 KHZ through U-bend
		818 tubes tested at 400 KHZ through first support
		834 tubes tested at 400 KHZ through sixth support
	Outlet	No tubes were inspected.

Results of Inspection

"A" Inlet

Row	Column	% Defect	Origin	Location
23	32	100	O.D.	6" from tube end
23	34	100	O.D.	8" from tube end
26	36	83	O.D.	5" from tube end
26	38	80	O.D.	8" from tube end
25	40	95	O.D.	12" from tube end
27	47	75	O.D.	19" from tube end

"A" Outlet

Row	Column	% Defect	Origin	Location
26	42	24	O.D.	2" above tubesheet
26	38	22	O.D.	3" above tubesheet
25	37	21	O.D.	3" above tubesheet
25	38	32	O.D.	2½" above tubesheet
24	43	21	O.D.	2" above tubesheet
24	39	21	O.D.	2" above tubesheet
23	41	21	O.D.	2" above tubesheet
23	42	23	O.D.	2" above tubesheet
22	33	29	O.D.	2" above tubesheet
19	31	21	O.D.	2" above tubesheet
19	32	21	O.D.	2" above tubesheet
19	36	21	O.D.	2" above tubesheet
24	53	22	O.D.	3" above tubesheet
26	53	23	O.D.	2" above tubesheet
27	54	21	O.D.	2" above tubesheet
26	54	26	O.D.	3" above tubesheet
25	54	21	O.D.	3" above tubesheet
24	54	21	O.D.	3" above tubesheet
24	55	21	O.D.	2" above tubesheet

"B" Inlet

Row	Column	% Defect	Origin	Location
45	41	86	O.D.	No. 1 support
22	38	76	O.D.	5" from tube end
31	41	61	O.D.	7" and 9" from tube end
21	36	90	O.D.	12" to 19" from tube end

"B" Outlet

No eddy current examinations were performed.

List of Plugged Tubes

"A" Steam Generator Row	Column	"B" Steam Generator Row	Column
23	32[1]	45	41[1]
23	34[1]	22	38[1]
26	36[1]	31	41[1]
26	38[1]	21	36[1]
25	40[1]		
27	47[1]		
1	8[2]		

[1] Defective tube (i. e., eddy current indication of 40% or above)

[2] Restricted tube, would not pass 0.650" probe

Weld Repair of Explosively Plugged Tubes

One previously exploded plug was weld-repaired in the "A" steam generator inlet at position R21C36.

5.2 Unit 2

Refueling 4 Inservice Inspection

Extent of Eddy Current Inspection

"A" SG	Inlet	162 tubes tested at 400 KHZ through U-bend
		925 tubes tested at 400 KHZ through first support
		248 tubes tested at 400 KHZ through sixth support
	Outlet	0 tubes tested at 400 KHZ through U-bend
		292 tubes tested at 400 KHZ through first support
		213 tubes tested at 400 KHZ through sixth support
"B" SG	Inlet	126 tubes tested at 400 KHZ through U-bend
		354 tubes tested at 400 KHZ through first support
		316 tubes tested at 400 KHZ through sixth support
	Outlet	0 tubes tested at 400 KHZ through U-bend
		639 tubes tested at 400 KHZ through first support
		430 tubes tested at 400 KHZ through sixth support

Results of Inspection

"A" Inlet

Row	Column	% Defect	Origin	Location
6	30	27	O.D.	Top of tubesheet
8	25	21	O.D.	Top of tubesheet
8	32	30	O.D.	Top of tubesheet
13	59	29	O.D.	½" above tubesheet
17	31	21	O.D.	½" above tubesheet
17	35	23	O.D.	1½" above tubesheet
17	37	36	O.D.	½" above tubesheet
17	42	27	O.D.	½" above tubesheet
18	35	26	O.D.	1" above tubesheet
18	36	26	O.D.	1" above tubesheet
18	37	24	O.D.	½" above tubesheet
18	38	30	O.D.	1" above tubesheet
18	43	27	O.D.	2" above tubesheet
18	54	21	O.D.	½" above tubesheet
20	42	26	O.D.	½" above tubesheet
20	46	31	O.D.	½" above tubesheet
20	47	31	O.D.	½" above tubesheet
20	50	29	O.D.	½" above tubesheet
20	52	21	O.D.	½" above tubesheet
21	34	22	O.D.	½" above tubesheet
21	46	21	O.D.	Top of tubesheet

"A" Inlet, continued ...

Row	Column	% Defect	Origin	Location
21	47	35	O.D.	½" above tubesheet
24	55	24	O.D.	½" above tubesheet
27	34	29	O.D.	Top of tubesheet
40	25	71	O.D.	5" above tubesheet

"A" Outlet

Row	Column	% Defect	Origin	Location
8	34	21	O.D.	1½" above tubesheet
10	39	29	O.D.	2" above tubesheet
10	40	22	O.D.	2" above tubesheet

"B" Inlet

Row	Column	% Defect	Origin	Location
14	38	29	O.D.	Top of tubesheet
18	41	21	O.D.	1" above tubesheet
19	42	36	O.D.	½" above tubesheet
21	34	31	O.D.	Top of tubesheet
23	58	37	O.D.	Top of tubesheet
25	51	36	O.D.	1" above tubesheet
26	41	22	O.D.	Top of tubesheet
27	27	35	O.D.	Top of tubesheet
29	54	22	O.D.	½" above tubesheet
32	40	29	O.D.	½" above tubesheet
32	42	39	O.D.	Top of tubesheet

"B" Outlet

Row	Column	% Defect	Origin	Location
7	26	21	O.D.	1" above tubesheet
9	26	21	O.D.	1" above tubesheet
10	26	22	O.D.	1½" above tubesheet
5	28	29	O.D.	1½" above tubesheet
9	28	24	O.D.	1" above tubesheet
18	29	27	O.D.	1" above tubesheet
7	30	22	O.D.	2" above tubesheet
12	30	26	O.D.	1" above tubesheet
9	31	22	O.D.	2" above tubesheet
13	31	27	O.D.	½" above tubesheet
18	32	29	O.D.	1" above tubesheet
9	33	23	O.D.	2" above tubesheet
12	33	29	O.D.	1½" above tubesheet
13	33	21	O.D.	1" above tubesheet
9	34	23	O.D.	1½" above tubesheet
16	35	25	O.D.	½" above tubesheet
7	36	21	O.D.	2" above tubesheet

"B" Outlet, continued ...

Row	Column	% Defect	Origin	Location
11	36	31	O.D.	2" above tubesheet
16	36	22	O.D.	½" above tubesheet
12	37	37	O.D.	1½" above tubesheet
15	37	23	O.D.	½" above tubesheet
8	38	24	O.D.	2" above tubesheet
4	39	23	O.D.	Top of tubesheet
11	39	26	O.D.	1" above tubesheet
13	39	27	O.D.	1" above tubesheet
14	39	26	O.D.	½" above tubesheet
10	40	24	O.D.	1" above tubesheet
13	40	29	O.D.	1" above tubesheet
9	41	21	O.D.	2" above tubesheet
13	41	31	O.D.	1" above tubesheet
12	42	26	O.D.	½" above tubesheet
13	42	21	O.D.	½" above tubesheet
10	43	22	O.D.	1½" above tubesheet
11	43	26	O.D.	1½" above tubesheet
13	43	21	O.D.	½" above tubesheet
10	44	31	O.D.	1" above tubesheet
11	44	34	O.D.	1" above tubesheet
10	45	23	O.D.	1½" above tubesheet
1[illegible]	46	34	O.D.	1" above tubesheet
9	47	21	O.D.	2" above tubesheet
10	47	21	O.D.	1½" above tubesheet
12	47	29	O.D.	½" above tubesheet
14	47	21	O.D.	½" above tubesheet
9	48	26	O.D.	1½" above tubesheet
15	48	2[illegible]	O.D.	½" above tubesheet
9	49	3[illegible]	O.D.	2" above tubesheet
10	49	2[illegible]	O.D.	1½" above tubesheet
11	49	21	O.D.	1" above tubesheet
12	49	21	O.D	1" above tubesheet
15	49	[illegible]1	O.D.	½" above tubesheet
18	49	29	O.D.	½" above tubesheet
11	50	29	O.D.	1½" above tubesheet
18	50	29	O.D.	1" above tubesheet
19	50	31	O.D.	1" above tubesheet
13	51	21	O.D.	1" above tubesheet
14	51	21	O.D.	½" above tubesheet
15	51	26	O.D.	1" above tubesheet
16	51	22	O.D.	½" above tubesheet
18	51	22	O.D.	1" above tubesheet
8	52	23	O.D.	2" above tubesheet
9	52	35	O.D.	2" above tubesheet
10	52	23	O.D.	2" above tubesheet
14	52	29	O.D.	1" above tubesheet
15	52	35	O.D.	1" above tubesheet
22	52	24	O.D.	½" above tubesheet
23	52	23	O.D.	½" above tubesheet
9	53	29	O.D.	2" above tubesheet

"B" Outlet, continued ...

Row	Column	% Defect	Origin	Location
14	53	22	O.D.	1" above tubesheet
15	53	27	O.D.	1" above tubesheet
16	53	21	O.D.	1" above tubesheet
19	53	26	O.D.	½" above tubesheet
22	53	23	O.D.	½" above tubesheet
7	54	21	O.D.	1" above tubesheet
9	54	32	O.D.	2" above tubesheet
10	54	21	O.D.	2" above tubesheet
14	54	22	O.D.	1" above tubesheet
15	54	34	O.D.	1" above tubesheet
23	54	21	O.D.	1" above tubesheet
9	55	21	O.D.	2" above tubesheet
15	55	21	O.D.	1" above tubesheet
16	55	21	O.D.	1" above tubesheet
23	55	22	O.D.	1" above tubesheet
13	56	21	O.D.	1" above tubesheet
15	56	21	O.D.	1" above tubesheet
18	56	29	O.D.	1" above tubesheet
22	56	21	O.D.	½" above tubesheet
25	56	22	O.D.	½" above tubesheet
11	57	21	O.D.	2" above tubesheet
14	57	24	O.D.	1" above tubesheet
18	57	23	O.D.	1" above tubesheet
19	57	21	O.D.	1" above tubesheet
22	57	21	O.D.	1" above tubesheet
23	57	23	O.D.	½" above tubesheet
7	59	23	O.D.	2" above tubesheet
11	59	21	O.D.	2" above tubesheet
15	59	35	O.D.	2" above tubesheet
9	60	22	O.D.	2" above tubesheet
16	60	29	O.D.	1" above tubesheet
9	61	21	O.D.	2" above tubesheet
11	61	27	O.D.	2" above tubesheet
13	61	29	O.D.	1½" above tubesheet
10	62	21	O.D.	1½" above tubesheet
13	62	23	O.D.	1½" above tubesheet
14	62	30	O.D.	1" above tubesheet
15	63	32	O.D.	1" above tubesheet
7	64	26	O.D.	2" above tubesheet
9	65	21	O.D.	1½" above tubesheet
15	65	21	O.D.	½" above tubesheet
23	65	23	O.D.	1" above tubesheet
15	67	23	O.D.	½" above tubesheet
19	67	22	O.D.	1½" above tubesheet
9	72	22	O.D.	1" above tubesheet

List of Plugged Tubes

"A" Steam Generator		"B" Steam Generator	
Row	Column	Row	Column
40	25[1]	No tubes plugged	
27	82[2]		

[1] Defective tube (i. e., eddy current indication of 40% or above).

[2] Did not show defect by eddy current testing, but was visually observed to be leaking.