NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

PDR

February 28, 1978

Mr J G Keppler, Director, Region III
Office of Inspection & Enforcement
U S Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Dear Mr Keppler:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Annual Report of Occupational Exposure
and Changes, Tests & Experiments
January 1 - December 31, 1977

In accordance with Appendix A Technical Specification 6.7.A.2 for the subject license, enclosed are two copies of the Annual Report of Occupational Exposure. The section on Changes, Tests and Experiments has been included in this report as a convenient means of meeting the annual reporting requirements of 10CFR50.59(b).

In our license amendment request dated October 31, 1977, which proposed to delete the requirement for an Annual Operating Report, we committed to submittal of a Narrative Summary of Operating Experience for the year 1977. The summary is contained in the attached report.

Yours very truly,

L.O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/deh

cc: Director, IE c/o Distribution Services Branch, DDC, ADM (40)
G Charnoff
MPCA
Attn: J W Ferman

Attachment

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## I. NARRATIVE SUMMARY OF OPERATING EXPERIENCE

1/1/77 to 2/22/77

Operated at 100% of rated power except for brief weekly reductions for control rod exercising and valve testing.

On 1/18/77 the accumulator on CRD hydraulic control unit 14-19 would not hold pressure due to stem packing leakage in the nitrogen charging valve. Evidence of packing material in the stem threads, and damage to the packing indicated improper packing installation. Instructions were issued concerning packing adjustments and accumulator operability was restored with the packing replaced on 1/18/77 (Reportable Occurrence No. M-RO-77-01).

On 1/25/77 it was noted that the SBGTS flow rate had not been recorded monthly in compliance with Technical Specifications which were issued 9/27/76. The test procedure was revised to provide for a monthly flow record and the administrative procedure for implementing new or revised Technical Specifications was also revised (Reportable Occurrence No. M-RO-77-02).

On 1/27/77 No. 3 TIP Ball Valve failed to close during routine operation of the TIP System. The valve was tapped and it closed. The cause of the failure could not be determined. The valve was replaced with the latest model, which was an improved operator (Reportable Occurrence No. M-RO-77-03).

2/23/77

A reactor scram occurred when a load rejection caused a turbine control valve fast closure. The load rejection resulted when a ice storm caused line problems which tripped the plant output breakers.

During plant startup a scram occurred due to high neutron flux when a reactor period of less than 5 seconds was obtained while withdrawing an in-sequence control rod. Analyses indicated that the combination of high temperature and high xenon concentration at the time of the occurrence established conditions such that criticality occurred on an unusually high reactivity worth control rod notch. The observed period was consistent with core analysis data (Reportable Occurrence No. M-RO-77-04). Following temperature reduction and xenon decay, plant restart was initiated.

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| 2/23/77 (Cont'd) | Administrative and Operating Procedures were subsequently revised and new core analysis procedures were instituted to identify high reactivity worth notches, place restrictions on their withdrawal, and clarify administrative procedures pertaining to anomalous reactivity changes. |
| 2/24/77 to 2/26/77 | Power was gradually increased to 100% of rated. |
| 2/27/77 to 3/17/77 | Operated at 100% of rated power except for brief weekly reductions for control rod exercising and valve testing.<br><br>On 3/1/77 following the installation of a redundant torus level transmitter, a discrepancy was noted between the two torus level indicators. Investigation revealed that the actual torus water volume was slightly below the minimum Technical Specification limit. Water volume was returned to the normal and the failed transmitter was replaced (Reportable Occurrence No. M-RO-77-05). |
| 3/18/77 | Commenced power reduction in preparation for scheduled maintenance shutdown. |
| 3/19/77 to 3/20/77 | Scheduled outage to perform the following maintenance:<br><br>a. Repaired pilot valve leakage on 4 reactor safety/relief valves and installed filters in the pressure sensing lines for all 8 valves.<br><br>b. Plugged leaking tube in low pressure feedwater heater 13B.<br><br>c. Replaced inboard seals on reactor feed pump #12.<br><br>d. Replaced 2 main condenser air vent valves. |
| 3/21/77 to 3/25/77 | Returned to power operation and increased power to 100% of rated.<br><br>On 3/21/77 the air-ejector sample system was found isolated, such that both air-ejector radiation monitors had been inoperable during startup. Operation of the monitors was re-established. The work control process and startup procedures were revised to prevent a recurrence. (Reportable Occurrence No. M-RO-77-06). |

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| 3/26/77 to 4/11/77 | Operated at 100% of rated power except for brief weekly reductions for control rod exercising and valve testing.<br><br>On 4/4/77 during a routine surveillance test, the set-point of one of the HPCI steam line area temperature switches was found to have drifted above the allowable Technical Specification limit. The switch was replaced (Reportable Occurrence No. M-RO-77-07). |
| 4/12/77 | Power was reduced to 66% of rated for load following. |
| 4/13/77 to 4/18/77 | Operated at 100% of rated power except for a brief reduction for control rod exercising and valve testing. |
| 4/19/77 | Following routine maintenance on #12 Reactor Protection MG Set, it was started and a transfer of load from the alternate source to the MG Set was attempted, initiating an expected Channel B half scram. The MG Set output circuit breaker had not been reset which caused a delay in the transfer. After approximately 1.5 seconds a reactor scram occurred due to a false indication of high flux on APRM #2. When the channel B power range neutron monitors were de-energized during the power source transfer, the shared LPRM inputs from APRM #6 were automatically removed from the APRM #2 averaging circuit. As a result of this action the LPRM average input increased, causing APRM #2 indication to increase above the scram setting. The MG Set transfer was completed and the plant was restarted. Procedures have now been prepared to calculate the effect on APRM's prior to making such transfers. An investigation of possible changes in LPRM assignment to minimize the affect on APRMs is in progress. |
| 4/20/77 to 4/22/77 | Power was gradually increased to 100% of rated. |
| 4/23/77 to 6/2/77 | Operated at 100% of rated power except for brief weekly reductions for control rod exercising and valve testing.<br><br>On 5/30/77, during the monthly RHR Motor Operated Valve Operability Test, "B" RHR Injection Valve MO-2013 failed to open. A line motor control center "open" contactor did not close as required when given the valve "open" signal causing the motor starter control power fuse to open. The fuse was replaced, the motor starter cleaned, and proper stroking and operation was demonstrated. (M-RO-77-09). |

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| 4/23/77 to 6/2/77 (Cont'd) | On 5/31/77, during the monthly RCIC Motor Operated Valve Operability Test, RCIC Outboard Steam Supply Isolation Valve MO-2076 failed to close. The main and limit switch gear train grease had deteriorated due to high ambient temperatures. Worn gears were replaced, the gear trains were cleaned and greased and valve operability was demonstrated. A ventilation modification has significantly lowered the ambient temperature. (M-RO-77-09). |
| 6/3/77 | Power was reduced to 92% of rated for load following. |
| 6/4/77 to 6/9/77 | Operated at 100% of rated power except for a brief weekly reduction for control rod exercising and valve testing. |
| 6/10/77 to 6/12/77 | Scheduled outage to perform operator licensing demonstrations, a CRD hydraulic return line isolation test and the following maintenance:<br>a. Plugged leaking tube in high pressure feedwater heater 14A.<br>b. Replace outboard seals on reactor feedwater pump #12.<br>c. Repacked miscellaneous valves. |
| 6/13/77 to 6/19/77 | Returned to power Operation. Increased power to 98% of rated. Power was limited to 98% of rated due to low feedwater pump suction pressure caused by leakage through the feed pump recirculation valves.<br><br>On 6/14/77 the torque switches for the RCIC steam line outboard isolation valve were improperly adjusted, such that the margin for normal deterioration was less than desired. The switches were subsequently properly adjusted and administrative procedures revised to clarify and improve control over such work. (M-RO-77-10).<br><br>On 6/17/77 a small leak was discovered in a welded joint on the 1" drain line connected to the "C" moisture separator drain line. The leak was temporarily patched. The original weld was found to be of poor quality. On 6/26/77, the original weld was ground off and the joint rewelded (M-RO-77-11). |
| 6/20/77 to 6/23/77 | Operated at 98% of rated power. |

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| 6/20/77 to 6/23/77 (Cont'd) | On 6/20/77, plant personnel were informed by General Electric Co. of an inappropriate assumption used in the determination of Cycle 5 MCPR limits. A conservative reanalysis increased the transient delta-CPR for all fuel by 0.08. At the time of the occurrence, the reactor was operating within the new limits (M-RO-77-12).<br><br>On 6/23/77, the flow through Standby Gas Treatment System was found to be below Technical Specification requirements. An isolation damper was found to be operating improperly due to a combination of normal wear and inproper installation of the air supply to the damper control. The valve controls were corrected and proper flow verified. (M-RO-77-13). |
| 6/24/77 | Power was reduced to 58% of rated due to high vibration on #11 Reactor Feedwater Pump (RFP). |
| 6/25/77 to 6/26/77 | Scheduled outage to repair feed pump recirculation valves CV-3489 and CV-3490 and #11 RFP. Inspection of the pump revealed damage to various components, including a severely rubbed shaft due to the suction flow guide being displaced, wiped inboard and outboard journal bearings, broken discharge and first stage diffuser capscrews, first stage diffuser vane to sideplate weld cracking, a broken first stage diffuser vane and a damaged impeller. |
| 6/27/77 to 6/30/77 | Returned to power operation. Power limited to 62% of rated pending completion of repair of #11 RFP.<br><br>On 6/27/77, during plant startup, the indication of the air ejector off gas radiation monitors was found to be abnormally low due to air leakage into the sampling system caused by an improperly open manual valve. The improper valve position was due to an incorrect valve identification tag. The valve was closed and valve tags were corrected. (M-RO-77-14). |
| 7/1/77 to 7/4/77 | Following repair of #11 RFP, power was gradually increased to 100% of rated. |
| 7/5/77 to 7/15/77 | Operated at 100% of rated power except for a brief reduction for control rod exercising, valve testing, and load following. |

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| 7/5/77 to 7/15/77 (Cont'd) | On 7/7/77, during routine operator inspection, a steam leak was observed at the 1-inch leak test connection to the RCIC steam supply line drain pot drain line. Inspection revealed poor quality of a welded joint. The leak was temporarily repaired with a clamp and packing. During the 1977 refueling outage, the faulty section of pipe and weld was replaced. (M-RO-77-15). |
| | On 7/9/77, the "A" recombiner train off-gas flow control valve (FCV-7489A) failed to stay closed after receiving a trip signal due to an accumulation of dirt in the associated solenoid valve. The valves operated properly after replacement of the solenoid valve internals. Previously installed supply line filters should prevent future dirt accumulation. (M-RO-77-16). |
| | On 7/11/77, following maintenance of the accumulator on CRD HCU 26-23, the nitrogen charging valve would not hold pressure. Inspection revealed that the valve was not properly seated in the instrumentation block, allowing nitrogen to leak past the seat and body O-rings. The O-rings were replaced and the valve was seated properly. (M-RO-77-17). |
| 7/16/77 to 7/19/77 | Reduced power to 56% of rated for weekly control rod exercising, valve testing and control rod pattern adjustments. Gradually returned power to 100% of rated. |
| | On 7/19/77, the nitrogen charging valve on CRD HCU 18-11 would not hold pressure. The nitrogen leaked past the valve stem as a result of defective packing, which had failed due to natural end-of-life. The valve was replaced. (M-RO-77-18). |
| 7/20/77 to 8/12/77 | Operated at 100% of rated power except for brief reductions for control rod exercising, valve testing, and load following. |
| | On 8/2/77, a small steam leak was discovered in a 45-degree elbow on the HPCI steam supply drain line to the condenser. Investigation revealed that a steam trap upstream of the elbow had failed causing erosion of the elbow. The leak was temporarily repaired with a clamp. During the 1977 refueling outage, the trap was repaired and the elbow was replaced. (M-RO-77-19). |
| | On 8/5/77, during the daily HPCI auxiliary oil pump test, a resistor in the HPCI governor control system failed resulting in a loss of DC power to the system. The resistor was replaced with an equivalent adjustable resistor. (M-RO-77-20). |

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| 7/20/77 to 8/12/77 (Cont'd) | On 8/10/77, the nitrogen charging valve on CRD HCU 30-11 would not hold pressure. The nitrogen leaked past the valve stem as a result of defective packing which had failed due to natural end-of-life. The packing was replaced. During the 1977 refueling outage, the packing was replaced in all 121 accumulator charging valves (M-RO-77-21). |
| 8/13/77 to 8/18/77 | Power was reduced to 54% of rated for control rod exercising, valve testing and to adjust the control rod pattern. Gradually increased power to 100% of rated. |
| 8/19/77 to 8/23/77 | Operated at 100% of rated power except for brief reductions for load following, control rod exercising and valve testing. |
| 8/24/77 to 8/25/77 | The recombiner system steam supply valve failed closed due to low air pressure caused by a leaking solenoid valve on the condensate demineralizer system and a partially plugged air filter. Reactor power was immediately reduced to 45% of rated. The valve was reopened by installing a temporary air supply line. The recombiner system and condenser vacuum were restored to normal. The solenoid valve was repaired and the filter was cleaned. As power was being increased, it was observed that the off-gas flow rate and recombiner bed temperature were low. Also, the SJAE off-gas radiation monitor readings were gradually increasing. A thorough investigation led to the conclusion that recombination was occurring at the air ejector after condensers. Normal operation was restored by shutting off the off gas flow to the air ejectors for a short time. It is believed that the trip of the recombiner steam supply valve allowed flame propagation from the recombiners back to the air ejectors. Power was returned to 100% of rated and then reduced to 93% for load following. |
| 8/26/77 | Operated at approximately 97% of rated power. Power coastdown due to end-of-cycle reactivity depletion in progress. |
| 8/27/77 | Electrical noise caused by a lightning storm initiated a trip of the recombiner trains. Power was reduced to 46% of rated until the recombiners were returned to operation and condenser vacuum was increased to normal. Power was then increased to 97% of rated. |

8/28/77 to 9/8/77

Operated at 92 to 96% of rated power except for brief reductions for valve tests, rod exercising and load following.

On 8/28/77, during the monthly RHR Motor Operated Valve Operability Test, "B" RHR Injection Valve (MO-2013) failed to open. The control leads for the "open" contactor for this valve are longer (about twice) than for any other contactor, resulting in a larger voltage loss. This, in conjunction with a slightly worn contactor, prevented operation. The contactor was cleaned and repaired, and control relays were installed to reduce control wire voltage loss. (M-RO-77-22)

On 9/3/77 all control rods were fully withdrawn.

9/9/77

Commenced scheduled outage to refuel the reactor and perform plant inspections, modifications and maintenance.

During operation of the RHR System, RHR torus cooling valve, MO-2009, failed to operate properly. Inspection revealed that the stem clamp set screws had sheared allowing the stem to rotate. A modified stem clamp utilizing a keyway was installed. (M-RO-77-23).

In addition to refueling, major items accomplished during the outage (9/9/77 to 11/10/77) included the following:

1) Type "A", "B" and "C" containment leak rate testing.
2) In-service inspection activities.
3) Surveillance tests and inspections.
4) CRD maintenance, including dye penetrant examination of collet housings.
5) Repair and repacking of miscellaneous valves.
6) Replacement of 10 LPRM strings.
7) Removal of 4 neutron sources and source holders from the core.
8) Capping of CRD hydraulic return line reactor vessel nozzle and drywell penetration and rerouting of return line to Reactor Water Cleanup System.
9) Installation of modified barrel assemblies and balance drums in both reactor feedwater pumps.
10) Preventive maintenance on all 8 safety/relief valves.
11) Replacement of 3 of 8 safety/relief valve discharge line ramsheads with T-quenchers.
12) Installation of 8-inch vacuum breakers on safety/relief valve discharge lines.
13) Replacement of feedwater low flow control valve CV 6-13 with a drag-type valve.

9/9/77 (Cont'd.)

14) Replacement of HPCI turbine steam isolation valve MO-2036.

15) Leak and eddy current testing of main condenser and feedwater heater tubes. Plugging of leaking and suspect tubes.

16) Inspection of turbine front standard, #3 and #4 stop and control valves, #2 and #4 CIV's and moisture separators.

17) Inspection of generator and exciter.

18) Modification of generator phase connection blocking.

19) Eddy current testing of hydrogen, exciter, stator and lube oil coolers.

20) Miscellaneous electrical inspections and maintenance.

21) Installation of new 480V load center.

22) Installation and maintenance of instrumentation in torus and drywell for relief valve discharge T-Quencher Tests.

23) Repair painting of drywell and torus interior skin.

24) Dredging of intake structure.

25) Modification of torus vent header supports to increase their load capability.

26) Installation of two additional condenser vacuum sensing lines to provide separation for condenser low vacuum scram sensors.

27) Machining of reactor vessel feedwater nozzles to remove stainless steel cladding and provide machined surfaces for thermal sleeves.

28) Installation of improved design feedwater spargers and thermal sleeves.

29) Inspection of 8 x 8 surveillance fuel bundle, reconstitution of the segmented test bundle, and inspection of selected fuel channels.

30) Installation of modified controls for turbine stop valve and control valve testing.

31) Completed installation of CW Pump flood protection trip circuitry.

32) Overhaul of both condensate pumps.

33) Miscellaneous maintenance and minor modifications.

9/11/77

During inspection of the drywell, an elbow and U-bolt of the "F" safety/relief valve discharge line were found damaged due to inadequate restraint. The elbow and U-bolt were replaced and an additional support was installed to reduce displacement of the line (M-RO-77-26).

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| 9/13/77 | Results of calculations using recently approved model changes were found to require slightly more restrictive fuel thermal limits at some exposures. (M-RO-77-25). A license amendment request was submitted to the NRC to revise the Technical Specifications. |
| 9/23/77 | Upon disassembly of RHR torus cooling valve, MO-2008, for stem replacement, the valve body seat ring threads were found to be stripped. Previous problems with shearing of the valve stem clamp set screws allowed the plug to rotate during attempts to operate the valve. Rotation of the plug unscrewed the seat which caused the threads to strip. The valve body was modified using a holding ring to position and secure the seat ring in place. (M-RO-77-27). |
| 9/25/77 | Core reloading was completed and preparations for feedwater nozzle work began. |
| 9/26/77 | A weekly IRM Rod Block Test was found to have not been performed. The surveillance file was revised to ensure identification of testing requirements associated with special plant conditions. (M-RO-77-28). |
| 10/2/77 | Three main steam line area temperature switches were found to trip slightly above the Technical Specification allowable setting. The switches were recalibrated. (M-RO-77-29). |
| 10/10/77 | During shutdown cooling operation, a fatigue crack was discovered on a "B" RHR loop relief valve 2-inch boss connection. The crack was ground out and the boss replaced with a weldolet. A restraint was installed. (M-RO-77-30). |
| 10/13/77 | Inspection of the torus internal catwalk revealed incorrect welded and bolted support attachment. All attachments were corrected to meet original construction requirements. (M-RO-77-31) |
| 10/30/77 | During a routine surveillance test the setpoint of a main steam line low pressure isolation switch was found to have drifted lower than allowed by Technical Specifications. The switch was recalibrated. (M-RO-77-32). |

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| 10/31/77 | Completed all CRD nozzle work.<br><br>During a surveillance test, the automatic transfer circuitry for Emergency Bus #15 did not function. Improper wiring of a new 480V load center, had resulted in a short circuit through the control fuses for the transfer circuit. The wiring was corrected and the fuses replaced. (M-RO-77-33). |
| 11/1/77 | Completed all feedwater nozzle and sparger work. |
| 11/5/77 | Completed reactor coolant leakage test. |
| 11/7/77 | Completed Type "A" primary containment integrated leak rate test. |
| 11/10/77 to 11/15/77 | Returned to power operation and increased power to 100% of rated. |
| 11/16/77 to 12/14/77 | Operated at 100% of rated power except for brief weekly reductions for control rod exercising and valve testing.<br><br>On 12/7/77, the secondary containment isolation dampers associated with reactor building vent supply unit, V-AH-4A, were found blocked by ice in the open position. Corrosion of the preheat coils inner steam-distribution tube resulted in the stagnation and freezing of the condensate within the coil causing the tube to rupture. The ice was thawed, the coil was repaired and the isolation dampers returned to service (M-RO-77-34). |
| 12/15/77 to 12/16/77 | Plant shutdown for scheduled outage to install and repair relief valve discharge T-Quencher test instrumentation and replace the topworks on "A" safety/relief valve. |
| 12/17/77 to 12/18/77 | Returned to power operation and increased power to 85% of rated. |

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| 12/19/77 to 12/22/77 | Conducted safety/relief valve discharge T-Quencher tests. Upon completion of the testing, power was returned to 100% of rated. |
| 12/23/77 to 12/31/77 | Operated at 100% of rated power except for brief weekly reductions for control rod exercising and valve testing. |

II NUMBER OF PERSONNEL AND MAN-REM BY WORK AND JOB FUNCTION

| WORK & JOB FUNCTION | NUMBER OF PERSONNEL (≥ 100 mrem) | | | TOTAL MAN-REM | | |
|---|---|---|---|---|---|---|
| | STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT WORKERS AND OTHERS | STATION EMPLOYEES | UTILITY EMPLOYEES | CONTRACT WORKERS AND OTHERS |
| REACTOR OPERATIONS & SURVEILLANCE | | | | | | |
| OPERATING PERSONNEL | 35 | 0 | 0 | 65.773 | 0.000 | 0.000 |
| HEALTH PHYSICS PERSONNEL | 8 | 0 | 0 | 15.415 | 0.000 | 0.000 |
| SUPERVISORY & ENGINEERING PERSONNEL | 28 | 11 | 19 | 19.803 | 6.662 | 6.973 |
| INSTRUMENT & CONTROLS PERSONNEL | 7 | 0 | 0 | 8.056 | 0.000 | 0.000 |
| ROUTINE MAINTENANCE | | | | | | |
| MAINTENANCE PERSONNEL | 31 | 57 | 1 | 51.963 | 28.871 | 0.220 |
| INSERVICE INSPECTION | | | | | | |
| HEALTH PHYSICS PERSONNEL | 0 | 0 | 1 | 0.000 | 0.000 | 0.447 |
| SUPERVISORY & ENGINEERING PERSONNEL | 0 | 2 | 3 | 0.000 | 0.515 | 3.766 |
| OPERATING PERSONNEL | 0 | 0 | 16 | 0.000 | 0.000 | 18.567 |
| *SPECIAL MAINTENANCE | | | | | | |
| MAINTENANCE PERSONNEL | 26 | 68 | 364 | 27.805 | 57.127 | 516.881 |
| HEALTH PHYSICS PERSONNEL | 4 | 0 | 26 | 1.760 | 0.000 | 31.365 |
| INSTRUMENT & CONTROLS PERSONNEL | 7 | 1 | 8 | 7.728 | 0.309 | 9.138 |
| WASTE PROCESSING | | | | | | |
| MAINTENANCE PERSONNEL | 11 | 0 | 0 | 5.157 | 0.000 | 0.000 |
| OPERATING PERSONNEL | 10 | 0 | 5 | 3.977 | 0.000 | 6.364 |
| SUPERVISORY & ENGINEERING PERSONNEL | 0 | 0 | 0 | 0.000 | 0.000 | 0.000 |
| REFUELING | | | | | | |
| MAINTENANCE PERSONNEL | 5 | 11 | 2 | 0.711 | 1.642 | 0.238 |
| OPERATING PERSONNEL | 20 | 0 | 3 | 4.161 | 0.000 | 0.390 |
| HEALTH PHYSICS PERSONNEL | 0 | 0 | 6 | 0.000 | 0.000 | 1.371 |
| SUPERVISORY & ENGINEERING PERSONNEL | 0 | 0 | 6 | 0.000 | 0.000 | 1.769 |
| SECURITY | 0 | 0 | 9 | 0.000 | 0.000 | 2.020 |
| **TOTAL | | | | | | |
| MAINTENANCE PERSONNEL | 73 | 136 | 367 | 85.636 | 87.640 | 517.339 |
| OPERATING PERSONNEL | 65 | 0 | 24 | 73.910 | 0.000 | 25.321 |
| HEALTH PHYSICS PERSONNEL | 12 | 0 | 33 | 17.175 | 0.000 | 33.184 |
| SUPERVISORY & ENGINEERING PERSONNEL | 28 | 13 | 28 | 19.803 | 7.178 | 12.508 |
| INSTRUMENT & CONTROL PERSONNEL | 14 | 1 | 8 | 15.784 | 0.309 | 9.138 |
| SECURITY | 0 | 0 | 9 | 0.000 | 0.000 | 2.020 |
| GRAND TOTAL: | 192 | 150 | 469 | 212.308 | 95.127 | 599.510 |

*DESCRIPTION:
1. Maintenance Performed in Primary Containment During Shutdown
2. Feedwater Sparger Modification
3. Torus Modifications
4. Reactor Building Crane Modifications

**INDIVIDUALS MAY BE LISTED UNDER MORE THAN ONE WORK AND JOB FUNCTION.

## III. CHANGES, TESTS AND EXPERIMENTS

The following sections include a brief description and a summary of the safety evaluation for those changes, tests and experiments which were carried out without prior NRC approval, pursuant to the requirements of 10CFR50.59(b).

### 1. REACTOR BUILDING MAIN STEAM CHASE EXHAUST FANS (ADDENDUM #2 to SRI 111)

#### Description of Change

A manual transfer of operation for main steam chase exhaust fans V-EF-24A and V-EF-24B was installed. The automatic transfer logic previously installed under SRI 111 was abandoned in place.

The original design provided for automatic switching to the alternate fan if failure occurred in the operating fan. Since that concept was presented, however, it was established that immediate fan switching is not necessary, and that the time available with a manual switching scheme is more than adequate for continued plant operation.

#### Summary of Safety Evaluation

Interlocks with SGTS and reactor building supply fans and Reactor building isolation capability are not affected by the modification.

### 2. INSTALLATION OF PILOT INLET FILTERS ON SAFETY/RELIEF VALVES (SRI 174)

#### Description of Change

Pilot inlet filters were installed on the 8 Target Rock safety/relief valves to maintain cleanliness of the pilot valves.

#### Summary of Safety Evaluation

The pilot filters had been removed previously because of concern that the filters could cause degradation of valve response time. However, drain and vent groove modifications to the valve eliminate the potential delay time problem. Testing has shown that a substantial amount of buildup in the filter does not affect valve performance.

3. DELETE ANNUAL REPLACEMENT REQUIREMENT FOR THE STACK FILTER (SRI 175)

Description of Change

The requirement for annual replacement of the HEPA stack filter has been deleted. The Stack HEPA filters will be retained in service as long as annual DOP testing verifies that they meet the filter efficiency testing requirements, the filter pressure drop limit is not exceeded and the recombiner system is not bypassed. In the event the recombiner system is bypassed the filter will be replaced after one year of service.

Summary of Safety Evaluation

FSAR Section 9.3.3.3 states that the annual filter unit replacement is based on the activity that would be released from an explosion in the off-gas filter after one year operation at the stack release rate of 100,000 uci/sec (after 30 minutes delay), which would result in 10% of the filter's activity being released to the environment. With the incorporation of the modified off-gas system, the hydrogen and oxygen (source of explosions) has been removed from the 30 minute delay pipe and the filter. Additionally, the off-gas is now filtered through 2 charcoal and 2 HEPA units prior to being compressed and stored in tanks before release.

4. LOAD CENTER ADDITION (75M094)

Description of Change

The new 480V load center was installed to reduce the loads carried by load centers B12, B14 and B23 and to provide capability for future requirements.

Summary of Safety-Evaluation

Only the non-essential loads of B12, B14 and B23 were transferred to the new load center. The new load center has an alternate supply from load center B2 which is supplied from the 4160 volt bus #14.

5. TORUS MODIFICATIONS (76M052)

Description of Change

The torus-to-support column connections were reinforced by adding weld metal to the existing web plate-to-shell weld, the lower wing plate-to-shell weld, the upper wing plate-to-shell weld, and the vertical stiffner-to-lower wing plate weld. Also, one inch parallel reinforcing plates were added on each side of the web plate.

Summary of Safety Evaluation

The modification was in accordance with ASME Code, Section III (through the Winter of 1975 Addenda). This modification increased the load carrying capability of the torus support structure.

6. RELOCATE HPCI GLAND CONDENSER RESTRICTING ORIFICE (77 M 012)

Description of Change

The HPCI gland seal condenser restricting orifice, RO-2058, was moved from downstream to upstream of the gland seal condenser. The purpose of the change is to reduce the pressure on the condenser during HPCI system startup and thereby minimize the potential for gasket extrusion from the shell-bonnet interface.

Summary of Safety Evaluation

System operating characteristics were not changed by this modification.

7. MODIFY TORUS VENT HEADER SUPPORTS (77 M 017)

Description of Change

The torus vent header support connections were reinforced by replacing the existing pins with high strength pins. The upper connection reinforcement clevis plates, with spacer plates, were bolted to the vent header collar and pinned to the support column. This modification increased the load carrying capacity of the support connections.

Summary of Safety Evaluation

This modification was performed in accordance with ASME Code, Section III, Subsection NF. The FSAR references the AISC Code as the applicable construction code. The ASME Code was developed from the AISC Code, therefore, an updated version of the same code was used.

8. REACTOR BUILDING CRANE (77Z019)

Description of Change

The originally installed reactor building crane trolley was replaced with a single-failure proof (redundant) trolley to reduce the probability of dropping a fuel shipping cask or other heavy load. The new trolley incorporates an 85 ton capacity dual redundant configuration main hoist and a 5 ton capacity conventional auxiliary hoist. The electrical controls were also modified to minimize single failure vulnerability.

Summary of Safety Evaluation

The structural and mechanical components were designed to meet the requirements of the governing codes and standards. All critical load bearing components of the main hoist have a minimum safety factor of 5.

9. INSTALLATION OF FEEDWATER CONTROL VALVE DIFFERENTIAL PRESSURE TAPS (77 M 021)

Description of Change

Pressure taps were installed on existing drain lines located upstream and downstream of the "A" Feedwater Control Valve to permit measurement of valve differential pressure.

Summary of Safety Evaluation

The modification was performed in accordance with the original design code, ANSI B.31.1, Power Piping.

10. LOAD MITIGATING SPARGERS (77 M 041)

Description of Change

The rams heads on the A, E and G safety/relief valve discharge lines were replaced with load mitigating spargers.

Summary of Safety Evaluation

Aspects of this modification which could conceivably affect the probability or consequences of an accident or malfunction previously analyzed were evaluated. The quencher is designed to result in an acceptable pressure drop and thereby eliminate feedback to the safety relief valve. Neither the safety relief valve nor nuclear steam supply system are affected by the modification.

Effects of the quencher on the piping has been specifically accounted for in the quencher and support design. The loads on containment with the existing ramshead were measured during relief valve test at Monticello during June, 1976, and the structural adequacy of the containment was demonstrated.

11. GEAR RATIO CHANGE ON RCIC OUTBOARD ISOLATION VALVE, MO-2076 (77 M 043)

The 50 to 1 worm and worm wheel of the Limitorque SMB-000 motor operator on MO-2076 were replaced by a worm and worm wheel having a 68 to 1 gear ratio due to wear observed on the original worm and worm wheel.

Summary of Safety Evaluation

The modification results in slower valve operating speed, but closing time remains within established limits. Valve component stresses as a result of this modification are acceptable.

12. RELIEF VALVE SOLENOID PLATES (77 M 046)

Description of Change

The leaking bellows test solenoid valves and air actuator solenoid valve for each safety/relief valve were relocated from a direct mounting on the valve to a plate which was located near the valve. The purpose of the modification is to simplify maintenance performed on the valves. The Prompt Relief Trip solenoid valves were removed from the solenoid valve group at this time.

Summary of Safety Evaluation

Relocation of the solenoid valves did not affect relief valve operation or bellows testability. At the present time there are no plans for using the Prompt Relief Trip System.

13. CYCLE 6 RELOAD (77 M 050)

Description of Change

The Monticello reactor core was changed for operation in CYCLE 6 by removing 132 fuel assemblies and replacing them with a like number having 2.62 w/o enrichment. The CYCLE 6 core configuration includes 100 MTB's (Reload 2), 48 GBH's (Reload 3), 204 LJ's (Reload 4), and 132 LJ's (Reload 5. All of the new fuel assemblies have finger springs.

Summary of Safety Evaluation

The Reload 5 Licensing Amendment prepared by General Electric Company NEDO-24032, contains sections which consider the mechanical design, nuclear characteristics, thermal-hydraulic analysis and safety analysis pertinent to the Reload 5 fuel added to the Monticello core for CYCLE 6 operation.

14. STR CYCLE 6 CHANGES (77 M 051)

Description of Change

Thirteen (13) irradiated segmented fuel rods were removed from the STR II fuel bundle, MTB 001. Seven (7) of these rods were replaced by unirradiated segmented rods containing a total of 28 segments; six (6) of the thirteen were replaced with rods having two (2) unirradiated segments each, or a total of 12. Thus, 40 new, unirradiated segments were added to the STR bundle.

Summary of Safety Evaluation

The safety evaluation contained in GE document NEDE-20179 with sections concerning Results from Design Evaluations, Fuel Operating and Development Experience, Nuclear Characteristics, and Safety Analysis for the reconstituted STR bundle, indicates that the changes made to MTB 001 should have no effect on the ability to operate the bundle and the core within all applicable thermal limits and safety considerations for CYCLE 6 operations.

15. REMOVAL OF SOURCES (77 M 056)

Description of Change

The four neutron sourceholders with sources were removed from the core at the end of CYCLE 5. (NOTE: The center core source had been removed in 1973, as previously reported.) This was done in conjunction with GE recommendations to preclude degradation of the sourceholders resulting from neutron embrittlement. The sourceholders and sources were not replaced for CYCLE 6 operation.

Summary of Safety Evaluation

For core average exposure above 8000 MWD/STU, sufficient neutrons are produced by fission product poisons to provide the required SRM countrate during shutdown. BOC-6 exposure was 8248 MWD/STU and is expected to be similar at the beginning of subsequent equilibrium cycles.

16. REMOVAL OF REACTOR VESSEL FEEDWATER NOZZLE CLADDING AND INSTALLATION OF SPARGERS (77 M 063)

Description of Changes

Stainless steel cladding, along with fatigued base metal, was removed from the reactor vessel feedwater nozzles by machining. Spargers incorporating a piston ring seal thermal sleeve were installed. These modifications were made to reduce the probability for feedwater nozzle crack initiation.

Summary of Safety Evaluation

The modification was performed in accordance with requirements of ASME Code, Section III, 1974 Edition with Addenda through Summer, 1976 and Section XI Edition with addenda through Summer, 1975

17. CRD RETURN ROUTED TO RWCU (77 M 065)

Description of Change

To eliminate the possibility of cracking in the stainless steel CRD hydraulic return line, the line was isolated and re-routed to the RWCU return line downstream of the last motor operated isolation valve. The new return line is constructed completely of carbon steel which is resistant to the type of cracking stainless steel is susceptable to. The isolation is maintained by two manually-operated stop valves.

Summary of Safety Evaluation

Operation of the CRD hydraulic system is in no way degraded by isolation of the return line. A special test proved that isolation did not affect any significant operating parameters, including normal rod movement, stall flow, settle time and pressure, exhaust water pressure and charging water pressure. Scram insercion time was also unaffected. The new return line was installed and tested in accordance with the applicable codes.

If the CRD pumps are required to provide coolant to the vessel, the two manual stop valves can be opened.

18. CAP CRD RETURN LINE NOZZLE (77 M 069)

Description of Change

To eliminate cracking of the CRD return line RPV nozzle, the nozzle was capped during the refueling outage. The drywell penetration was also capped and all return line piping in the drywell was removed. The reactor nozzle cap is four-inch Sch. 120 ASTM-A-182, Gr. 316 with a 0.02 percent carbon maximum, while the containment vessel cap is six-inch Sch. 80 ASTM-A-350, Gr. LF1.

Summary of Safety Evaluation

These modifications do not create the possibility of a new accident, increase the probability or consequences of a previously analyzed accident or decrease the margin of safety for any Technical Specification. The caps were installed and tested in accordance with the applicable codes.

19. REPLACEMENT OF DRYWELL CIM/CAM WITH A PARTICULATE CAM (77 M 096)

Description of Change

The drywell CIM/CAM was replaced by a particulate CAM. This was done because of unreliable operation of the drywell CIM/CAM. The CAM uses the same sample inlet and discharge connections that the CIM/CAM used.

Summary of Safety Evaluation

The replacement of the drywell CIM/CAM with a particulate CAM did not create any new potential accident considerations. Iodine monitoring of the Drywell atmosphere is not required or deemed necessary.

20. RELOAD FUEL ASSEMBLY (5S) LJ 3736 FOR CYCLE 5 OPERATION (75 M 087)

Semi-Annual Operating Report Number 10 reported that fuel bundle LJ 3736 contained 14 tabs on the water/spacer capture rod rather than the customary 7 tabs. Visual observation in September, 1977, confirmed that LJ 3736 does in fact contain only 7 tabs.