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NARRATIVE SUMMARY OF OPERATING EXPERIENCE FOR 1977; REPORT OF FACILITY CHANGES, TESTS OR EXPERIMENTS.

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NMP-0725

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

February 27, 1978

Mr. Boyce H. Grier
Director
United States Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, PA. 19406

RECEIVED
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U. S. NUCLEAR REGULATORY COMMISSION
Mail Section

RE: Docket No. 50-220
DPR-63

Dear Mr. Grier:

In accordance with your letter dated September 22, 1977 from Mr. George Lear to Mr. G.K. Rhode, we are submitting the following reports only in place of the former Annual Operating Report:

1. Narrative Summary of Operating Experience for 1977
2. Report of Facility Changes, Tests or Experiments pursuant to 10 CFR 50.59(b)

One copy of the Annual Tabulation of Occupational Exposure Report and 10 CFR 20.407 Report (copy enclosed) is being submitted under separate cover directly to the Office of Inspection and Enforcement.

The requirement for an Annual Operating Report was deleted by Amendment 20 to our Operating License DPR-63 dated January 3, 1977.

Very truly yours,

ORIGINAL SIGNED BY R.R. SCHNEIDER

R.R. Schneider
Vice President -
Electric Production

mtm

Enclosures (2 copies)

xc: Director, Office of I&E (40 copies)

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NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION

I. NARRATIVE SUMMARY OF OPERATING EXPERIENCE FOR 1977

A summary of major operating events and major safety-related maintenance for the period January through December 1977 is submitted in the following monthly narratives. Operating statistics are included in the previously submitted monthly reports.

1. January

The unit was operated for 744 hours during the month of January, with a unit availability factor of 100% and a unit capacity factor of 94.4%. The second stage reheaters were out of service during the entire month due to excessive steam leaks.

Jan. 1 - Jan. 7

The reactor was operated at approximately 97% of rated producing approximately 603 mwe.

Jan. 8

At 0530 load was reduced in order to repair and rebrush #11 Reactor Recirculating Pump M.G. set.

Jan. 9

Work was completed on #11 Reactor Recirculating Pump M.G. set and load was increased via recirculation as PCIOMR limits allowed.

Jan. 10

Unit output at approximately 600 mwe.

Jan. 11 - Jan. 31

The unit was operated at approximately 600 mwe for the remainder of the month.

2. February

The event was operated for 646 hours during the month of February, with a unit availability factor of 96% and a unit capacity factor of 87%. The 2nd stage reheaters were out of service during the entire month of February.

Feb. 1 - Feb. 4

The unit was derated approximately 40 mwe due to the unavailability of the second stage reheaters and core thermal considerations.

Feb. 5

At 0218 load was reduced to 485 mwe in order to remove #15 Reactor Recirculating pump MG set from service for rebrushing. At 0320 load was reduced to 400 mwe in order to adjust control pattern.

Feb. 6

Maintenance was completed on #15 Reactor Recirculating pump MG set and load was increased as PCIOMR limits allowed.

Feb. 7

Continued load increases via recirculation flow to approximately 600 mwe.

Feb. 8 - Feb. 22

Unit load was maintained at approximately 605 mwe during this time via small changes in recirculation flow.

Feb. 23

At 0508 the reactor scrammed due to a turbine trip caused by a carbon dust build-up on the field leads under the generator.

Feb. 24

The cause of the trip was investigated and repaired. At 2353 the reactor was made critical and unit start-up was in progress.

Feb. 25

The unit was back on line and load was increased during the day via rod withdrawals and recirculation increases.

Feb. 26 - Feb. 28

Unit output was increased via recirculation flow to approximately 603 mwe and the unit continued to operate at this load for the remainder of the month.

3. March

The unit was operated for 98.7 hours during the month of March. The annual maintenance overhaul and refueling outage started on March 5 and continued for the remainder of the month.

March 1 - March 3

The unit was operated at approximately 605 mwe. The second stage reheaters were out of service during this period.

March 4

At 2100 load was reduced to 360 mwe for scheduled outage.

March 5

At 0239 the unit was removed from service. At 0815 all rods were fully inserted and a reactor cool-down continued for the remainder of the day.

March 6 - March 31

The scheduled outage continued for the remainder of the month of March. The major work performed involved:

a) The replacement of 160 fuel assemblies
b) Replacement of 4 feedwater spargers
c) Turbine and generator overhaul and inspection
d) CRD Overhauls
e) In-service inspection
f) Local Leak Rate Testing of isolation valves & penetrations
g) Preventive Maintenance
h) Plant Modifications

Highlights of the major outage work accomplished were as follows:

a) The in-core sipping operation was discontinued on March 17, 1977 due to malfunctioning of the in-core sipper head. Since greater than 85% of the fuel assemblies scheduled for re-insertion into the core for Cycle 5 had been tested (all of the improved fuel design) with no leaker fuel assemblies detected, it was felt that the probability of any leaker fuel assemblies in the remaining population was very small.

   A total of 108 control rod cells were tested with acceptable results. The twelve peipheral fuel assemblies, scheduled for discharge, were not sipped. Analysis of the sipping data identified eight fuel assemblies as having perforated cladding.

b) Twenty-five CRDs were overhauled and replaced during the scheduled control rod drive maintenance, found linear indications on three (3) collet.

3. March (Cont.)

c) The replacement of 19 LPRMs was completed by March 26.

d) Work on replacement of reactor head safe ends commenced on March 17.

e) During routine ultrasonic examination of cleanup system piping inside the primary containment, reportable indications were found on cold bends 25N and 25-S. Piping sections were scheduled for removal and replacement.

f) Retainer tubes in control rod drive mechanisms were removed from the reactor vessel. Assemblies were replaced.

Turbine:

a) Generator Inspection completed and reassembly in progress.

b) Reheaters & Moisture separators inspected.

c) Three TCIVs, two TSVs, two TCVs and three TBVs were disassemblied and inspected.

4. April

The unit was unavailable for service during the entire month of April due to continuation of the Annual Outage.

April 1 - April 30

Major work completed during this period was as follows:

Reactor:

a) The existing feedwater spargers were removed. Numerous dye penetrant indications were found in the clad on the southeast nozzle. Machining of the clad was performed.

b) Five reactor head safe ends have been completed, three others have welding completed and seven others are in various stages of completion.

c) Installation of new replacement clean-up system piping is in progress:

Turbine:

a) Generator reassembly was completed.

b) Final PT work on reheaters completed.

4. April (Cont.)

c) The new Philadelphia gear cluth for #13 feedwater pump was aligned and reassembly was in progress.

d) Turbine valve assembly completed.

5. May

The unit was unavailable for service during the entire month of May due to the continuation of the Annual Overhaul.

May 1 - May 31

Major work done during this period was as follows:

Reactor:

a) The installation of two spargers were completed. The remaining two spargers were in the process of final fit-up and installation.

b) Installation of clean-up piping was nearing completion.

c) Reactor safe end work completed.

d) It was decided to replace the control rod drive nozzle on the reactor vessel. Delivery of a new safe end and thermal sleeve expected early in June.

e) In-service Inspection of the emergency condenser safe ends revealed a small crack in the east nozzle. A new elbow was ordered as a replacement.

f) All drywell snubbers were inspected and four were found to be inoperable.

Turbine:

a) Turbine work completed except for replacement of 1 moisture separator cone.

b) Final assembly of #13 feedwater pump was nearing completion.

c) Inspection of #11 feedwater pump was completed.

6. June

The unit was unavailable for service during the entire month of June due to the continuation of the Annual Overhaul.

6. June (Cont.)

June 1 - June 30

Major work done during this period was as follows:

Reactor:

a) Feedwater sparger replacement was completed.

b) Control rod hydraulic nozzle on the return to the reactor was replaced. During preplanned inservice inspection of reactor vessel emergency condenser steam outlet nozzle, an unacceptable linear PT indication was found in the pipe at the safe end to pipe weld joint. In grinding out this indication, it was necessary to grind through the entire pipe wall. The 90° pipe elbow was removed and replaced.

c) Two plugs were installed in this nozzle to allow repairs while water level was raised to accommodate refueling.

d) Installation of clean-up system piping outside the drywell was completed except for one weld that will be done in parallel with work on the system inside the drywell.

e) Replacement of 160 fuel assemblies was started on June 17 and completed on June 24.

Turbine:

a) Turbine work has been completed.

7. July

The annual overhaul and refueling outage was completed on July 10, 1977. The unit availability factor for July was 43.8% with a unit capacity factor of 34.4%.

July 1 - July 9

Final outage wrap-up in progress. During hydrostatic test found leak at weld between emergency condenser valves 39-01 and 39-03. Weld was ground out and repaired. Both PT and Radiographic examinations were made during and following repairs. The defect was found in the valve material upon metallographic examination of a boat sample.

July 10

At 1330 the mode switch was placed in start-up and criticality achieved at 1625.

7. July (Cont.)

July 11

The reactor was critical but a hold-up was incurred due to problems with the MSIV test circuit.

July 12

During routine unit startup, #11 Emergency Cooling System was discovered to be inoperable. Ultra-sonic testing of system 11's condensate return inlet valve (39-01) indicated that the valve stem was separated from the disc. The normally open valve was run to closed position and a surveillance test was run on the redundant Emergency Cooling System (#12) in accordance with Technical Specification. No problems were encountered. A Technical Specification change was requested to allow normal operation with one Emergency Cooling System inoperable, beyond the seven days currently allowed. In addition, it was requested that the increased surveillance for operation with an inoperable emergency cooling system be performed on a weekly rather than a daily basis.

At the conclusion of Cycle 5, maintenance will be performed on the inoperable emergency cooling system to place it back in service.

July 13

During routine startup operations, 13 CRDs were discovered to have scram times greater than allowed by Technical Specification. These were identified during a refueling operator surveillance test. The reactor was placed in a hot shutdown condition as required by Tech. Specs.

The ASCO scram pilot valves failed to operate correctly because the pressure diaphrams and exhaust diaphrams were cracked or torn. A total of 26 pilot valves (2 per CRD) were repaired by replacing the above two diaphrams in each. In addition, the five body gaskets per valve, which fit between the valve body and the pilot body and the bonnets and the valve body were replaced.

Following testing of the electromatic relief valves at 1312 it was necessary to further delay the startup to make repairs to the #122 electromatic relief valve.

July 14

At 1119 the mode switch was placed in startup, and at 1230 the reactor was critical. All rods met scram time requirements.

7. July (Cont.)

July 15

At 0102 the mode switch was placed in run. At 0140 #122 electromatic relief valve was retested and the reactor was shutdown again to repair #122 electromatic relief valve. At 0356, mode switch to startup and all rods in at 0542.

July 16

Final repairs to #122 electromatic relief valve were made. A defective valve seat was found and relapped.

July 17

At 1708, the mode switch was placed in startup, and at 1750 the reactor was critical.

July 18

At 0327, the mode switch was placed in run. The electromatic reliefs were tested at 0500 and were found acceptable. At 0900 the turbine was rolled and the generator was synchronized at 1007. The unit load was increased via rod withdrawal.

July 19 - July 31

Continuing to increase load via rod withdrawal and recirculation flow increases.

8. August

The unit was operated for 744 hours during the month of August with a unit availability factor of 100% and a unit capacity factor of 95.1%.

Aug. 1 - Aug. 31

During this time numerous small load reductions and increases (≅40 mwe) were made due to condensate demineralizer changeouts.

9. September

The unit was operated for 720 hours during the month of September with a unit availability factor of 100% and a unit capacity factor of 97%.

Sept. 1 - Sept. 30

During this month load was reduced three times to allow for condensate demineralizer changeouts. Reactor power was maintained between 1830 and 1840 mwth. This level of reactor power coupled with the improvements made during the annual overhaul to the vapor lines to the steam jet air ejectors allowed the unit to achieve its highest generation September output since start of operation.

10. October

The unit was operated for 594.25 hours during the month of October with a unit availability factor 79.8% and a unit capacity factor of 71.3%

Oct. 1

Load was reduced for a control rod interchange. Rod interchange completed at 1000. Physics calculation took the remainder of the day.

Oct. 2

Increasing load via recirculation flow increases as thermal limits allowed.

Oct. 3 - Oct. 5

Second stage reheaters in service. Increasing load via recirculation flow increases.

Oct. 6

Core thermal power at 1804 mwth, generator at 600 mwe. Continuing load increases.

Oct. 10

Core thermal power at 1821 mwth, generator load at 610 mwe.

Oct. 21

Core thermal power at 1827 mwth, generator load at 614 mwe. At 1712, started a unit shutdown due to high leakage into the drywell sumps and for snubber inspection.

At 2222 tripped turbine.

Oct. 22

At 0028 the mode switch was placed in startup. All rods in at 0312.

Oct. 23 - Oct. 25

Maintenance replacing reactor recirculating pump seals.

Oct. 26

Commenced a reactor startup and the reactor was critical at 0705. Reactor scrammed due to an IRM instrument spike. The reactor was made critical again at 1040. At 1400 it was discovered that #123 electromatic relief valve was leaking. A cut valve gasket was found and replaced. At 1815, started to insert rods. All rods in at 2038.

10. <u>October</u> (Cont.)

Oct. 27

With #123 electromatic relief valve repaired, commenced a reactor startup at 1802.

Oct. 28

At 0300, the mode switch was placed in run. The unit was synchronized to the line at 0530. Load increasing via rod withdrawal and recirculation increases.

Oct. 29

Continued with load increases to 530 mwe.

Oct. 30

Continued load increases. At 2400 unit load at 1811 mwth with a generator load of 605 mwe.

Oct. 31

Load reduced to 508 mwe for a condensate demineralizer change at 2100. Load increased to 590 mwe at 2400.

11. <u>November</u>

The unit was operated for 556.05 hours during the month of November, with a unit availability factor of 77% and a unit capacity factor of 74%.

Nov. 1

Increasing generator load to 618 mwe via recirculation flow increases.

Nov. 6

The unit was at 1831 mwth and 617 mwe. At 0730 started to reduce load due to high leakage into the drywell sumps. Mode switch to startup at 1247 and all rods in at 1512.

Nov. 7 - Nov. 10

The unit was off line in order to replace seals on #11, #14 and #15 reactor recirculation pumps.

Nov. 11

Commencing a reactor startup. The reactor was critical at 1630. Mode switch placed in the run mode at 2325.

11. <u>November</u> (Cont.)

Nov. 12

Unit on line at 0157. Increasing load via rod withdrawals and recirculation flow increases.

Nov. 13

Continued increasing load via rod withdrawals and recirculation flow.

Nov. 14

Continued load increase to 614 mwe.

Nov. 15

Unit load at 618 mwe.

Nov. 16

Reduced load at 0116 to change over a condensate demineralizer. Load increased to 622 mwe.

Nov. 17 - Nov. 26

Unit operated in a steady state condition at ≅620 mwe.

Nov. 27

While testing TSVs, turbine tripped at 2208, causing a reactor scram.

Nov. 28

The cause of the trip was investigated, and a failed air solenoid valve was found on #11 TSV. The reactor was made critical at 1220. The mode switch was placed in run at 1540. At 1650 turbine - generator on line, and increasing output via rod withdrawals.

Nov. 29 - Nov. 30

Continued to increase load via rod pulls and recirculation flow increases. Unit at ≅594 mwe at 2400 on November 30.

12. <u>December</u>

The unit was on line for 744 hours during the month of December, with a unit availability factor of 100% and a unit capacity factor of 97.6%. The second stage reheaters were in service during the entire month.

Dec. 1 - Dec. 2

Continued to increase recirculation flow as PCIOMR limits allowed to 1838 MWTH and 622 MWE.

Dec. 3 - Dec. 16

Continued steady state operation at ≈ 1830 MWTH and 620 MWE.

Dec. 17

At 2019 reduced load to 550 MWE via recirculation flow for a condensate demineralizer change out.

Dec. 18

By 0000 load had been increased to 615 MWE and load was increased to 625 MWE by 0600.

Dec. 19 - Dec. 22

Continued steady state operation at ≈ 1840 MWTH and 625 MWE.

Dec. 23 - Dec. 25

Load varied during this period due to power demands.

Dec. 26

At 0145 unit load was 591 MWE. Increased power via recirculation flow to 613 MWE by 0830.

Dec. 27 - Dec. 29

Increased load to 627 MWE and continued steady state operation.

Dec. 30

Started a load reduction at 2130 for a control rod interchange. At 2350 attained minimum recirculation flow.

Dec. 31

Completed rod sequence exchange at 0600. Increase load via recirculation flow during the remainder of the day as PCIOMR limits allowed. Unit produced highest monthly net generation (442,822 MWH) in the history of plant operation.

II MODIFICATIONS INCORPORATED DURING 1977

Summaries of modifications made to the station during 1977 are submitted in the following narratives. Those modifications applicable to controls established within the scope of 10CFR50.59 and Appendix B to 10CFR50 are described as well as other station improvements which reflect increased reliability efforts. All applicable provisions of 10CFR50.59 have been complied with. The required safety analysis have been reviewed by the Site Operations Review Committee and by the Safety Review and Audit Board. It has been concluded that none of the changes which have been made constitute an unreviewed safety question. A notation is included in the description where the modification was made subsequent to NRC approval and/or change in Technical Specifications.

N1.76.18 - This non nuclear safety related plant improvement concerns itself with adding a file area in the Administration building basement for Document Control and Archiving utilizing a computer based information retrieval system. Major order 4813 began in the fall of 1976 and was completed in May of 1977. Compliance to Document Control is in accordance with ANSI Standard N.45.2.9 and NRC Regulatory Guide 1.88.

N1.76.15 - This modification consisted of replacing 2 Crosby safety relief valves in the feedwater system with 2 Crosby valves with improved design features. These valves were 2- 3/4" x 1" JR-WR-B models set at 600 psig with 10% overpressure and designed for ASME Section III code requirements. The Safety Evaluation for this change considered it not to be an unreviewed safety question as defined in 10CFR50.59 (a) (2).

N1.76.14 - The expansion of the security building and perimeter fencing was accomplished this past year. This included increased office and storage area, training area, photo identification, maintenance area and expanded security measures. This was a non NRC Safety related modification to the site.

N1.76.03 - This modification consisted of installing adjustable stops to 15 non-return valves in the steam lines to the feedwater heaters. This is intended to prevent the clapper portion of the valves from lodging in one position. These adjustable stops will also provide a remote means of freeing the clapper in the event it did become lodged. The safety evaluation for this change concluded it not to be unreviewed safety question as defined in 10CFR50.59 (a) (2).

N1.77.28 and N1.75.09 - These were combined modifications to the Emergency Vent System in the control room. This modification consisted of adding additional length and width to provide 7.5 duct diameters which is free from bends and other disturbances. Test ports will be added to facilitate pitot tube testing in both the control room and Rx Building Emergency Vent lines. These changes will be made in accordance with ANSI N.510-1975 and ACGIH Industrial Vent Standards. The Safety Evaluation concluded it not to constitute any unreviewed safety question as defined in 10CFR50.59 (a) (2).

N1.75.44 - Adding temperature monitors in the Torus was accomplished during the past outage. This provides redundant measurement of suppression chamber water temperatures. One RTD was installed but five more will be done as time permits. An RTD transmitter and a display recorder will be added shortly. This is considered reliability related only.

N1.77.26 and N1.77.27 - Several improvements to the new off gas system were performed during the refueling outage. These improvements included: 1) installing sample valves and filters on the sample supply to the hydrogen analyzer. 2) installing a low point drain connection at the stack. 3) replaced the orifice plate on the steam diluted off gas flow to the recombiners. 4) replace root valves with gate valves in system 77.4 5) add a flow nozzle at a location after the recombiner condenser and vent cooler. 6) modify the seat disc on a valve at the vacuum pump inlet connection. 7) install a new off gas grab sample at the outlet of the second stage steam jet air ejectors and 8) install a high point vent connection in a dead line to the standby recombiner to the 1" test connection at the inlet to the active recombiner. All off gas improvements increase the reliability of the present system.

N1.77.35 - This modification required adding test valves on the emergency condenser steam line, east and west, for use in the testing of isolation valve leakage. This modification complies with Appendix B 10CFR50 criteria and ASME code Section II. The safety analysis concluded it did not constitute an unreviewed safety question as defined in 10CFR50.59 (a) (2). These test points will be used during the ILRT.

N1.77.37 - A modification to the feedwater sampling system was performed during the refueling outage. The existing feedwater sample probe was replaced with a low cobalt bearing root valve assembly. Previously, the presence of cobalt bearing alloys in the original valve and tubing masked the desired cobalt measurements. General Electric performed the installation and their safety analysis reflected the mod did not constitute any unreviewed safety questions in accordance with 10CFR50.59 and Appendix B to 10CFR50 criteria.

N1.77.38 - Feedwater flow straighteners were removed from the feedwater lines during inspection of the feedwater nozzles. Removal of the flow straighteners does not affect the safety functions of the high pressure feedwater system. The safety evaluation considered criteria of 10CFR50.59 and Appendix B 10CFR50 and concluded that this modification did not constitute a unreviewed safety question.

N1.75.03 - This modification concerned itself with adding a main condenser water box priming vacuum pump. Originally condenser water boxes were primed by using mechanical vacuum pump hoggers and the prime was maintained during operation by operating the steam jet air ejector system. This required a connection between the main steam system and the circulating water system. This modification eliminates the interconnection and the possibility of main steam condensibles entering the circulating water system. The two mechanical vacuum pumps added will isolate the cooling water from the main steam.

N1.77.31 - A modification to change the source of water to the control rod drive hydraulic system from the condensate surge and storage tanks to the condensate demineralizer was made during the refuel outage. It was desired to have a supply of higher quality water to the CRD pumps on a normal usage basis. Additional check valves and a cut off line were added to the existing system to provide this new source of water. The safety evaluation concluded that the design, procurement and installation of the pressure retaining components of this modification would comply with Appendix B 10CFR50 and that it did not constitute an unreviewed safety question as defined in 10CFR50.59 (a) (2).

N1.72.06 - Number 13 feedwater pump clutch was redesigned and installed during the refueling outage. The old clutch was continually requiring repair. A newly designed disc/dental type clutch was added. Initial engagement of this clutch will be by oil actuated discs similar to the previous design. When the input and output shafts are syncronized a dental clutch is engaged for a more positive connection. This change was consisered a reliability related change to the system and is not safety related.

N1.74.15 - This addition consisted of installing a wet automatic fire protection system to cover the low level laboratory at floor elevation 261' and a wet system to cover the railroad car unloading dock at elevation 261' in the reactor building track bay extension. This will provide a means to counter any gasoline spills in the truck area.

N1.73.01 - A new 125 ton redundant crane was tested early in 1977 wrapping up the installation of the hoist for the reactor building. This replaced the original hoist that was there. NRC approval of this modification accurred on 12/10/75. The new crane can prevent dropping heavy loads in the event a cable or other critical parts of the hoisting equipment fail. Loads will be spent fuel casks, reactor head and shield plugs. This modification was reviewed by several groups prior to installation with conclusions that compliance to 10CFR50.59 and Appendix B 10CFR50 was fulfilled.

N1.77.04 - This modification consisted of replacing 1 amp 50 mv shunts in the turbine stop valve circuitry (PRS 11&12) with 10 ohm 25 w precision resistors. Previous methods for testing the condition of switch contacts on these valve sets. The safety evaluation concluded that the modification of this nature to the reactor protection systems improves the reliability of the system by making it easier to test pursuant to IEEE Stnd. 279 and that it did not constitute any unreviewed safety question as defined in 10CFR50.59 (a) (2).

N1.77.48 - This modification to the instrument air compressor lines concerned itself with the addition of an isolation valve on the pressure instrumentation line between the compressor and the receiver tank. Use of the isolation valve allows one to calibrate the pressure instrumentation with the unit in operation. The review of the safety evaluation considers the implications of 10CFR50.59 and Appendix B 10CFR50 and these regulations are complied with.

N1.77.57 - The reactor building extension was modified in December 1977 when the Peele door seal was modified. The modification to this door was to replace the existing removable seals with longer seals. The change in design allows easier

N1.77.57 (cont.) - access to the reactor building extension by rubber tired vehicles. Since the reactor building extension door becomes part of the secondary containment Appendix B to 10CFR50 applies. This modification does not constitute an unreviewed safety question as defined in 10CFR50.59 (a) (2).

N1.77.07 - Prior to the refueling outage the lifting frame for the reactor cavity to the internals storage pit shield plugs was redesigned to improve the lifting capability of the frame. At the same time the redesign allows for the shield plugs to be removed from the frame while underwater. Previously, the studs holding the frame to the shield plug were not accessible because they were located underneath the shackle bolt pins. The new design has the shackle pin split and the stud holding the frame to the shield plug is between the split shackle pins. When the shackle swings clear the stud is readily accessible for removal. Appendix B to 10CFR50 has been followed. The modification does not constitute an unreviewed safety question as defined in 10CFR50.59 (a)(2).

N1.77.22 - This modification consisted of adding under voltage (low voltage) relays to powerboards 102 & 103 as required by NRC edict to preclude postulated degraded grid voltage conditions. Four under voltage relays with inverse time and fast reset characteristics were added. This modification did not constitute an unreviewed safety question as in compliance with 10CFR50.59 (a) (2).

N1.76.11, N1.77.23 and N1.77.39 - These are modifications to the plant which are all related to the clean up system. N1.76.11 concerned itself with improving the seismic control equipment located just outside the isolation valves. N1.77.23 concerned itself with replacing stainless steel piping and hangars outside the drywell with carbon steel to alleviate pipe cracking problems which have required unit shutdowns in the past. Both N1.76.11 and N1.77.23 are considered non class I items because these portions exist outside the isolation valves. N1.77.39 concerned itself with replacing 22 feet of 6" stainless steel 304 piping with carbon steel piping inside the drywell. This was to eliminate cracking caused by fatigue on the cold bent portions of the piping. Appendix B to 10CFR50 and 10CFR50.59 were complied with in this modification.

N1.76.19 - This modification involved replacing 18 reactor head safety valve and one reactor head vent valve safe ends between the reactor head connections and flanges where the valves are attached. Since cracks had developed in the existing safe ends the replacement was with non sensitized 304 stainless steel which is highly resistant to stress corrosion cracking. Appendix B 10CFR50 and 10CFR50.59 have been complied with.

N1.77.12 - The major modification during the refueling outage was the work on the feedwater sparger and nozzles. The full interference fit between the thermal sleeve and the feedwater nozzle was changed to an elliptical fit. Loss of interference with an elliptical fit is not expected since the elliptical fit remains elastic even after a few thermal cycles. The elliptical fit provides excellent mechanical support. The safety evaluation for this modification reviewed all aspects in accordance with Appendix B 10CFR50 and compliance with 10CFR50.59 has been fulfilled.

N1.77.24 - During the refueling outage the control switch circuitry for containment spray pumps 111 and 112 was modified to eliminate a poor design condition. The circuitry was a logic "and" and did not permit running of either pump should one be out of service. The modification consisted of rewiring the controlling switches to all containment spray pump circuits in parallel so that a logic "or" is present. This allows the isolation of either pump should one be taken out of service. 10CFR50 Appendix B and 10CFR50.59 have been complied with. The safety evaluation determined that this modification does not constitute any unreviewed safety questions.

N1.72.03 - The Containment Air Dilution System was added to the plant safeguards systems during the past year. In the improbable event of a LOCA the CAD system would suppress released hydrogen to below flammability limits. Appendix B to 10CFR50 and 10CFR50.59 have been complied with in this major modification. The NRC approved this addition on December 26, 1974. As a part of this project, the valves on channel 11 & 12 RPS were separated from a common power supply by the addition of another power supply to provide redundancy. Override will be placed on these valves such that they can be manually reopened after initiation of the RPS closure signal.

N1.76.22 - The reactor building and turbine building crane controls were modified prior to the refueling outage to improve the reliability of the control scheme. Improvements included replacing fuses for better protection of SCR's and adding meters, test parts and board test equipment. The safety evaluation noted that the function of the circuitry involved was reliability related and did not affect the safety implications as noted by 10CFR50.59 or Appendix B 10CFR50.