DONALD C. COOK NUCLEAR PLANT
1994 ANNUAL OPERATING REPORT

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COMPILED BY: William R. Moran
W. R. Moran
Senior Engineer
American Electric Power Service Corporation

REVIEWED BY: L. S. Gibson
L. S. Gibson
Assistant Plant Manager
Cook Nuclear Plant

APPROVED BY: E. E. Fitzpatrick
E. E. Fitzpatrick
Vice President
Indiana Michigan Power Company

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd.)

1.0 INTRODUCTION

1.1 PLANT DESCRIPTION

The Donald C. Cook Nuclear Plant is owned by Indiana Michigan Power Company and is located five miles north of Bridgman, Michigan. The plant consists of two nuclear power units, each employing a Westinghouse pressurized water reactor nuclear steam supply system. Each reactor unit employs an ice condenser reactor containment system. The American Electric Power Service Corporation was the architect-engineer and constructor.

Unit 1 and 2 reactor design power outputs (and licensed rating) are 3250 Mwt and 3411 Mwt, respectively. Unit 1 approximate gross and net electrical outputs are 1056 Mwe and 1020 Mwe, respectively. Unit 2 approximate gross and net electrical outputs are 1100 Mwe and 1060 Mwe, respectively. The main condenser cooling method is open cycle using Lake Michigan water as the cooling source for each unit.

1.2 REPORT PREPARATION

This report was compiled by W. R. Moran with the following individuals contributing information as follows:

D. L. Noble	-	personnel exposure summary
K. R. Worthington	-	steam generator ISI summary
C. C. Savitscus	-	changes to procedures
C. C. Savitscus	-	tests or experiments not described in the FSAR
R. S. Ptacek	-	challenges to pressurizer PORVs and safety valves
S. W. McLea	-	reactor coolant specific activity
T. A. Georgantis	-	results of irradiated fuel inspections
C. C. Savitscus	-	changes to facility - RFCs, MMs, PMs, and TMs

2.0 PERSONNEL RADIATION EXPOSURE SUMMARY

Table 1 provides a summary of the number of station, utility, and contractor (and others) personnel receiving exposures greater than 100 millirem in 1994. The total record dose for all personnel was 484.653 rem as measured by thermoluminescent dosimetry (TLD) and reported in accordance with Regulatory Guide 1.16.

TABLE 1 - ANNUAL OPERATING REPORT - RG 1.16 FOR 1994

	# PERSONNEL >100 Mr			TOTAL MAN-REM		
	STAT.	UTIL.	CONT.	STATION	UTILITY	CONTRACT
Reactor Operations & Surveillance						
Maintenance personnel	0001	0000	0011	000.172	000.000	002.069
Operations personnel	0048	0001	0002	011.558	000.338	000.336
Health physics personnel	0018	0000	0029	003.805	000.000	004.354
Supervisory personnel	0001	0000	0000	000.176	000.000	000.000
Engineering personnel	0003	0000	0000	001.001	000.000	000.000
Routine Maintenance						
Maintenance personnel	0115	0001	0403	053.600	000.295	167.607
Operations personnel	0026	0001	0040	006.470	000.413	012.361
Health physics personnel	0023	0000	0076	010.986	000.000	028.420
Supervisory personnel	0002	0000	0000	000.885	000.000	000.000
Engineering personnel	0017	0000	0006	005.134	000.000	001.660
In-Service Inspection						
Maintenance personnel	0002	0000	0073	000.254	000.000	027.899
Operations personnel	0004	0000	0018	000.876	000.000	007.119
Health physics personnel	0000	0000	0008	000.000	000.000	003.122
Supervisory personnel	0000	0000	0000	000.000	000.000	000.000
Engineering personnel	0002	0000	0005	000.252	000.000	003.413
Special Maintenance						
Maintenance personnel	0001	0001	0106	000.119	000.113	035.800
Operations personnel	0002	0000	0017	000.995	000.000	006.201
Health physics personnel	0003	0000	0000	000.380	000.000	000.000
Supervisory personnel	0000	0000	0000	000.000	000.000	000.000
Engineering personnel	0000	0002	0000	000.000	000.377	000.000
Waste Processing						
Maintenance personnel	0000	0000	0002	000.000	000.000	001.293
Operations personnel	0000	0000	0002	000.000	000.000	001.701
Health physics personnel	0002	0000	0017	000.855	000.000	003.848
Supervisory personnel	0000	0000	0000	000.000	000.000	000.000
Engineering personnel	0000	0000	0000	000.000	000.000	000.000
Refueling						
Maintenance personnel	0003	0000	0030	000.833	000.000	007.493
Operations personnel	0008	0000	0025	003.727	000.000	011.323
Health physics personnel	0007	0000	0016	002.064	000.000	002.620
Supervisory personnel	0000	0000	0000	000.000	000.000	000.000
Engineering personnel	0000	0000	0000	000.000	000.000	000.000
TOTALS						
Maintenance personnel	0116	0002	0520	054.978	000.408	242.161
Operations personnel	0076	0002	0082	023.626	000.751	039.041
Health physics personnel	0029	0000	0127	018.090	000.000	042.364
Supervisory personnel	0002	0000	0000	001.061	000.000	000.000
Engineering personnel	0019	0002	0011	006.387	000.377	005.073
GRAND TOTALS	0242	0006	0740	104.142	001.536	328.639

3.0 STEAM GENERATOR IN-SERVICE INSPECTION

3.1 Unit 1 Inspection Summary

Eddy current inspection of the Unit 1 steam generators (S/Gs) was performed during February and March 1994. The scope of this effort is listed in Sections I through III of the attachment to this report.

3.2 Unit 2 Inspection Summary

Eddy current inspection of the Unit 2 steam generators was performed during October of 1994. Approximately 6.5% of the total number of tubes in S/Gs 22 and 23 were inspected with an eddy current bobbin coil probe in accordance with Technical Specifications. In addition, due to observed tube damage in the tube lanes of S/Gs 22 and 23, 108 tubes in each generator were inspected from 3" above the flow distribution baffle (FDB) to the end of the tube on the hot and cold legs. Tube damage was attributed to mechanical interaction between the tubes and equipment used to perform pressure pulse cleaning (PPC) on the secondary side of the steam generators. A summary of the inspection scope is listed in Table 2.

TABLE 2 - Summary Unit 2 Steam Generator Inspection Scope

Description	S/G 22	S/G 23	TOTAL
Inspected Full Length	67	67	134
Inspected from the seventh support plate on the cold leg (7C) to tube end hot (TEH) (Selected at random)	154	154	308
Low Row Tubes Inspected 7C to TEH	14	14	28
Tubes inspected FDB+3" to tube end hot and tube end cold in the tube lane region	108	108	216
TOTAL	343	343	686

No imperfections were found as a result of the Technical Specification eddy current inspection. However, based on the results of the special eddy current maintenance inspection for mechanically damaged tubes, a total of three tubes in S/G 22 and six tubes in S/G 23 were plugged and removed from service. None of the tubes inspected during the special eddy current maintenance inspection exhibited signs of service induced degradation. A listing of the tubes plugged and removed from service is listed in Table 3.

TABLE 3 - Summary Unit 2 Steam Generator Tube Repairs

S/G 22	S/G 23
Row 1 Column 92	Row 1 Column 5 Row 1 Column 92
Row 1 Column 93	Row 1 Column 6 Row 1 Column 93
Row 1 Column 94	Row 1 Column 7 Row 1 Column 94

4.0 CHANGES TO PROCEDURES

This section contains a brief description of the procedure changes implemented under the provisions of 10CFR50.59 and the associated safety evaluations.

4.1 OPERATIONS HEAD PROCEDURES

4.1.1 Plant Cooldown from Hot Standby to Cold Shutdown

Description of Change:

01 OHP 4021.001.004 revision 23 changed the Unit 1 upper low temperature over pressurization protection (LTOP) enable temperature value from 331°F to 324°F.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that technical justification for the Unit 1 upper LTOP enable temperature of 324°F exists. The temperature 331°F was conservatively used in the past for operator convenience to eliminate confusion between the units.

4.1.2 Filling and Venting the Reactor Coolant System

Description of Change:

02 OHP 4021.002.001 change sheet 4 changed the Unit 2 upper LTOP enable temperature value from 331°F to 300°F. However, chapter 4 of the Update Final Safety Analysis Report (UFSAR) references a Unit 2 LTOP enable temperature of "331°F". This setpoint change in the UFSAR upper LTOP enable temperature also impacts the following procedures:

2 OHP 4021.001.001, Rev. 15, CS-5
2 OHP 4021.001.004, Rev. 15, CS-1
2 OHP 4021.008.001, Rev. 4, CS-5
2 OHP 4021.008.002, Rev. 9, CS-1
2 OHP 4021.008.003, Rev. 5, CS-4
2 OHP 4021.017.002, Rev. 8, CS-1
2 OHP 4021.017.003, Rev. 4, CS-4
2 OHP 4030.STP.004E, Rev. 2, CS-1
2 OHP 4030.STP.004W, Rev. 2, CS-1
2 OHP 4030.STP.051N, Rev. 5, CS-3
2 OHP 4030.STP.051S, Rev. 5, CS-3
2 OHP 4021.017.003, Rev. 4, CS-3
2 OHP 4030.STP.030, Rev. 19, CS-5

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that technical justification for setting Unit 2 upper LTOP enable temperature at 300°F was provided.

4.1.3 Operation of the Essential Service Water (ESW) System

Description of Change:

12 OHP 4021.019.001 change sheet 7 allowed for the Unit 1 east ESW header to be placed in service with a non-standard system lineup. Typically, the header is placed in service by filling the header with the cross-tie to the Unit 2 west ESW header. Because of the design change that replaced 1-WMO-707, the cross-tie piping is not connected and therefore it is necessary to fill the header by an alternate means. The header will be filled by hooking up a fire hose between manual drain valves 2-ESW-183 and 1-ESW-181.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the reactor was defueled while the Unit 1 east ESW header was inoperable and the Unit 2 west ESW header was declared inoperable during this evolution thus the procedure is within the allowances of the technical specifications, which allow 72 hours for an ESW header to be inoperable. It is also noted that an operator was stationed near the drain valves to shut the valves in the event it became necessary, and that the hose was restrained to protect against hose whip.

4.2 PLANT MANAGER PROCEDURES

4.2.1 The Control and Processing of Contaminated Water Generated in the Secondary System

Description of Change:

12 PMP 6010 RPP.700 revision 0 provided guidelines for handling and processing contaminated water in the secondary system, as a result of a steam generator tube rupture or major tube leak.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the procedure will not change the process for review and approval of proposed changes to water processing systems.

4.2.2 Update of the Meteorological Information Tables

Description of Change:

12 PMP 6010 OSD.001 revision 8 updated meteorological information tables (X/Q and D/Q valves), and the distance to the nearest milk producing animal.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that this change updates the meteorological data that is used to calculate doses from routine releases, the distance to the nearest milk producing animal, and the background data used in meteorological information and dispersion assessment system calculations. This updated information does not adversely impact equipment important to safety and does not increase the probability of an accident.

4.2.3 Internal Exposure Monitoring for Radioactive Material

Description of Change:

12 PMP 6010 RPP.200, revision 4, was updated to delete requirements that were redundant in other procedures and to clarify the Radiation Protection Department's expectations from the passive monitoring program. The procedure does not include routine in vivo bioassays as written in section 11.4.8 of the UFSAR. We are deleting the requirement requiring routine in vivo bioassay, but will continue to do them to establish baselines and to investigate possible uptakes of radioactive material.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the present program will exceed the requirements of the regulations. The revision requires whole body counts before receiving dosimetry, upon termination of employment and as needed for investigational purposes. In addition, the present surface contaminated monitors that all personnel are required to pass through before exiting the restricted area will detect internal contamination below the levels required by the regulations.

4.3 TECHNICAL HEAD PROCEDURES

4.3.1 Collection of Radiological Effluent Monitoring Program (REMP) Surface Water Samples

Description of Change:

12 THP 6010 RPP.630 revision 0 specified locations in the Offsite Dose Calculation Manual and table 3.12-1 of the Technical Specifications that were different from those in the UFSAR and the Emergency Plan. These documents are to be updated.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the revision to the procedure does not adversely impact any safety equipment and the monitoring program supplements the REMP by verifying that the measurable concentration of radioactive materials and levels of radiation are not higher than expected.

4.3.2 Collection of Milk Samples

Description of Change:

12 THP 6010 RPP.635, revision 0, a new procedure, replaced 12 THP 6010 ENV.054. The procedure included a change to indicate that one of the farms used for the collection of milk is no longer in the dairy business. This is a change to the UFSAR figure 2.7-3 and to the Emergency Plan figure 12-18.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the intent of UFSAR section 2.7 is satisfied. The radiological environmental monitoring program measures the effect that routine and inadvertent releases of radioactive material have on the environment.

4.3.3 Procedures Changed Due to New 10CFR Part 20

Description of Change:

The new 10CFR Part 20 changed the methodology of determining doses. Procedures 12 THP 6010 RPP.205, Rev. 3, 12 THP 6010 RPP.405, Rev. 3, 12 THP 6010 RPP.406, Rev. 5, 12 THP 6010 RPP.418, Rev. 1, 12 THP 6010 RPP.405, Rev. 4, 12 THP 6010 RPP.418, Rev. 2, were changed to meet the intent of the new 10CFR Part 20.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that only the method of determining doses was changed because of the new 10CFR Part 20. These procedure changes do not introduce a new accident, did not increase the probability of an accident and did not increase the consequence of any accident.

5.0 TESTS OR EXPERIMENTS NOT DESCRIBED IN THE FSAR

This section describes procedures classified as "Tests and Experiment," implemented under the provisions of 10CFR50.50, including the associated safety evaluation.

5.1 TESTS

5.1.1 Post Maintenance Testing of the Main Steam Isolation Dump Valves

Description of Change:

**12 IHP 6030 IMP.030, revision 5, change sheet 6, allowed for the installation of a mechanical jumper between the process test connections (MPI-211, MPI-212 through MPI-241, and MPI-242) on the steam generator main steam isolation valve (MSIV) dump valve lines downstream of the three way motor operated test valve. The purpose of the mechanical jumper is to allow for the pressurization of the steam line downstream of the three way valve to verify that the dump valve is seating properly prior to realigning the steam path from the MSIV to the dump valve following maintenance. Performing post maintenance testing utilizing this method is expected to prevent the inadvertent closure of the MSIV following dump valve maintenance in the event that the dump valve stroke is not properly set, resulting in dump valve seat leakage.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the installation of the temporary mechanical jumper will not affect the ability of the available main steam isolation valve dump valve to perform its design function (i.e., open on a main steam isolation signal to facilitate rapid MSIV closure). Additionally, all equipment in the steam generator MSIV compartment required for safe shutdown is high energy line break and environmentally qualified, therefore, in the event that the mechanical jumper failed, the ability to safely shut down the unit would not be affected.

5.1.2 Analog Rod Position Indication (ARPI) Prototype Testing

Description of Change:

**12 EHP SP.057 allowed for data to be taken during testing on the ARPI system. The data assisted with the development of electronics that are being used for the prototype ARPI upgrade project (RFC-DC-12-3119). During the performance of the test, one or two ARPI positions were connected to the new APRI data processing electronics. Control room panel indications including the rod bottom alarm were not active for these two rods in the control bank D.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the ARPI does not have any safety functions; the safety functions will not be bypassed; routine operating procedures will continue to be followed; the operator will be aware of the rod position at all times; the test will be performed while the reactor is subcritical and with adequate shutdown margin.

5.1.3 Fuel Rod Hi-Magnification/Eddy Current Examination

Description of Change:

**12 EHP SP.068 allowed for the examination of fuel rods as part of the effort to determine the root cause of fuel failures at Cook Nuclear Plant.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the overall objective of the procedure is to retain the fuel assembly functional and structural properties consistent with its pre-inspection condition; only one fuel assembly at a time was loaded into the system performing the inspection; there is no mechanism for the procedure to affect any other portions of the plant because the inspection/ reconstitution will be performed in the fuel transfer canal and spent fuel pool; and dropping a fuel assembly from the elevator is bounded by the existing fuel handling accident.

6.0 CHALLENGES TO PRESSURIZER POWER OPERATED RELIEF VALVES AND SAFETY VALVES

During 1994, there were no challenges on either Unit 1 or Unit 2 to the pressurizer power operated relief valves or the pressurizer safety valves as a result of the valves being called upon to mitigate an actual overpressure condition.

7.0 REACTOR COOLANT SPECIFIC ACTIVITY

During 1994, there were no instances on either Unit 1 or Unit 2 in which the reactor coolant I-131 specific activity exceeded the limits of Technical Specification 3.4.8.

8.0 IRRADIATED FUEL EXAMINATIONS

8.1 IN-MAST FUEL SIPPING (IMFS)

IMFS technology was used for the first time at Cook Nuclear Plant to detect fuel failures during 1994. During the refueling outages of both units, all 193 fuel assemblies were inspected using IMFS technology. Modifications were made to the manipulator crane in each containment to accommodate the IMFS system. Westinghouse Electric Corporation was contracted to perform these modifications.

To explain how the IMFS system detects fuel failures, during core offload, when a fuel assembly is raised out of the core, the reduction of water pressure due to the change of depth of water allows fission gases in failed fuel rods to expand. When the fuel assembly is raised into the manipulator crane mast to the full up position, a trickle flow of dry air is pumped through nozzles located at the bottom opening of the mast. This air rises through the mast, stripping any fission gases that may be present on the surface of the fuel rods. After this air reaches the surface of the refueling cavity, still within the mast, this offgas air is directed to a detector that measures radioactivity. Increases in radioactivity above background levels indicate the presence of at least one failed fuel rod in that fuel assembly. IMFS cannot determine which rod is failed, only that the assembly contains at least one or no failed fuel rods.

During the Unit 1 1994 refueling outage, twelve fuel assemblies were identified as containing failed fuel rods, and three other fuel assemblies were suspected of containing at least one failed fuel rod.

During the Unit 2 1994 refueling outage, three fuel assemblies were identified as containing failed fuel rods, and one other fuel assembly was suspected of containing at least one failed fuel rod.

8.2 VISUAL EXAMINATIONS

During the core offload for both units during 1994, all fuel assemblies were inspected per procedures **12 THP 6040 PER.353 and 12 SHP 4050 QC.002. In accordance with these procedures, each assembly is inspected using binoculars. During the transit of each fuel assembly to the spent fuel pool during core unload, the fuel assembly is inspected on all four sides. The examiner is looking specifically for torn or missing grid straps, missing or damaged fuel rods, excessive clad hydriding, or rod bow to gap closure. This inspection is primarily intended to detect fuel damage caused by mechanical interaction between fuel assemblies or baffle jetting, and is done each refueling. There was no indication of any fuel damage during these inspections in 1994.

8.3 ULTRASONIC TESTING (UT)

Prior to introducing IMFS technology to Cook Nuclear Plant, detection of fuel failures was primarily performed using UT technology. Now that IMFS technology is used at Cook Nuclear Plant, UT is now used as a confirmatory measurement, and to identify which fuel rods in the fuel assembly are failed. Typically, all irradiated fuel assemblies scheduled for reload are examined by UT. Also, any fuel assemblies found to be failed or suspect by IMFS are examined by UT. UT is performed in the spent fuel pool after core unload and prior to core reload. The goal of the combined use of IMFS and UT technologies is to prevent the reload of leaking fuel into either core at Cook Nuclear Plant.

UT works by a probe transceiver sending a high frequency sound wave into a fuel pin and measuring the strength of the returning signal, or "ring back". If there is water present in the fuel rod, the amplitude of the "ring back" will be diminished in relative comparison to fuel rods that do not contain water. If a fuel rod contains water, it is extremely likely that it is a failed fuel rod. The probe transceiver is inserted horizontally into the fuel assembly at an axial location just above the bottom grid strap. The inspection continues for each fuel assembly until the probe is pushed past each fuel rod.

During 1994, UT services for Cook Nuclear Plant were contracted to Westinghouse.

During the 1994 Unit 1 refueling outage, 135 fuel assemblies were inspected using UT technology. The results of the Unit 1 UT exam are as follows:

- Of the fuel assemblies that were inspected by IMFS and found not to be failed, no leaking fuel rods were found.
- Of the three fuel assemblies identified by IMFS to be suspect, one was found to contain a failed fuel rod; in the other two fuel assemblies, no leaking fuel rods were found.
- Of the twelve fuel assemblies identified by IMFS to contain at least one failed fuel rod, thirteen failed rods were found in ten of the fuel assemblies; no failed rods could be found in the other two fuel assemblies.

During the 1994 Unit 2 refueling outage, 121 fuel assemblies were inspected using UT technology. The results of the Unit 2 UT exam are as follows:

- Of the fuel assemblies that were inspected by IMFS and found not to be failed, no leaking fuel rods were found.
- Of the one fuel assembly identified by IMFS to be suspect, no leaking fuel rods were found.
- Of the three fuel assemblies identified by IMFS to contain at least one failed fuel rod, eight failed rods were found (at least one in each assembly).
- In both refueling outages, no failed fuel rods were reloaded into the core.

8.4 HIGH-MAGNIFICATION VIDEO EXAMINATION OF PERIPHERAL FAILED RODS

During August 1994, prior to the Unit 2 refueling outage, an examination was performed on selected fuel assemblies that were identified by UT during previous fuel leak detection campaigns to contain failed rods in the outer two rows of the fuel assembly. The purpose of the examination was to obtain information to help determine the root cause of the occurrence of fuel failures at Cook Nuclear Plant. Several of the rods inspected were in fuel assemblies identified during the Unit 1 refueling outage.

The inspection was performed using high-magnification video cameras and tooling that would lift or lower, and rotate selected fuel rods within the fuel assembly skeleton envelope.

This inspection was contracted to Westinghouse. Siemens Power Corporation personnel were present to witness the inspection. Fuel fabricated by each fuel vendor was inspected. 52 rods in 24 fuel assemblies were inspected.

The inspection revealed that grid-to-rod fretting was occurring randomly in selected fuel rods and selected mid-grid locations. Typically, only one rod per fuel assembly was noted to have this wear. No definitive pattern to where the wear was occurring was found, although a preference to fuel assemblies that had resident time in a core location next to the baffle was noted. Indications of baffle jetting or debris fretting wear were not observed. No other fuel failure mechanisms were identified.

8.5 INTRUSIVE FAILED FUEL VIDEO/FIBERSCOPE EXAMINATIONS

After the Unit 2 refueling outage, in November and December 1994, an intrusive failed fuel inspection of failed fuel was performed at Cook Nuclear Plant. This inspection was performed to obtain additional information with regard to the root cause of the occurrence of fuel failures at Cook Nuclear Plant. Fuel rods identified by UT as being failed and located toward the center, not on the periphery, of the fuel assembly were inspected, to determine if there were any similarities to the results found in the August inspection.

This inspection was contracted to Westinghouse. Only fuel assemblies manufactured by Westinghouse were inspected. Siemens has been contracted to perform a similar inspection in 1995 on fuel assemblies manufactured by Siemens.

During the Westinghouse inspection, 13 failed and 27 non-failed rods were inspected in 9 fuel assemblies. Selected fuel assemblies, one at a time, were placed in a special inspection elevator installed in the fuel transfer canal specifically for this inspection. The fuel assembly was inverted in the elevator, the bottom nozzle removed, and, one at a time, selected fuel rods were removed for inspection. Each rod was inspected using a high-magnification video camera. Each failed rod was then tested by passing it through an encircling eddy current coil. The cell location from which the failed rod was removed was also inspected using a fiberscope camera to observe potential damage to grid straps, springs, dimples and adjoining fuel rods, if any. Non-failed fuel rods were returned to the fuel assembly skeleton to the location from which it came; failed fuel rods were placed in a special rod storage basket that is stored in a spent fuel pool rack storage cell location. In two instances, stainless steel pins were placed in fuel assembly skeleton locations in place of the failed rod. This may allow reuse of these fuel assemblies should it be decided to use reconstituted fuel. Upon completion of the testing plan for each fuel assembly, the bottom nozzle was reinstalled, the fuel assembly was reinverted to its upright position and returned to its spent fuel pool storage rack cell location.

Again, grid-to-rod fretting was observed on most of the failed rods and on two of the non-failed rods that were inspected. The fretting indications were through-wall in some locations, and were not through-wall in other locations. The fiberscope inspection revealed damage occurring to selected dimples and springs on grids corresponding to locations with indications of failed fuel. Also again, no pattern fuel assembly wear has been found, although the preference to its occurrence in fuel assemblies with resident time in the reactor core next to the baffle was reconfirmed.

9.0 CHANGES TO FACILITY

This section contains a brief description of the design changes implemented under the provisions of 10CFR50.59 and the associated safety evaluations.

9.1 DESIGN CHANGES (RFCs)

9.1.1 Replace Residual Heat Removal (RHR) System Valves

Description of Change:

RFC-DC-12-1978 allowed for the replacement of Unit 1 RHR heat exchanger outlet valves IRV-310 and IRV-320 and the Unit 1 RHR heat exchanger bypass valve IRV-311. The butterfly valves were changed to ball type.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that this change has an insignificant impact on the loss of coolant accident (LOCA) analysis. The new ball valves have better flow control characteristics as compared to the old butterfly valves and so were to alleviate erratic control problems the operators were experiencing. The fail safe position and the seismic qualifications of the valves remain unchanged.

9.1.2 Reactor Protection and Control Process Instrumentation Replacement Project

Description of Change:

RFC-DC-12-2985 revision 0 was written to allow the installation of Taylor MOD 30 equipment in Unit 1 and Unit 2 reactor protection, engineered safety systems, reactor control, and water intrumentation systems. However, design deficiencies were discovered during factory acceptance testing and RFC-DC-12-2985 revision 1 was written to allow the installation of Foxboro "SPEC 200" and "SPEC 200 MICRO" equipment in lieu of Taylor Mod 30 equipment in the reactor protection and engineered safety systems and Taylor MOD 30 equipment in the non-safety related reactor control and water instrumentation systems.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the reactor protection/engineered safety systems are not being changed, only the method in which they are being met (digitally instead of analog). However, since the NRC's position, as elaborated in their draft Generic Letter, is that this changeout involves an unreviewed safety question, we have supplied the NRC with the necessary documentation for their review. This modification was implemented after NRC approval.

9.1.3 Volume Control Tank (VCT) Level Instrumentation

Description of Change:

RFC-DC-12-3057 revision 0 allowed for the installation of isolation and test valves in the sensing lines of the VCT level instrumentation. These valves allow one transmitter to be isolated when necessary, without taking the other out of service. Revision 1 reduced the scope of the original design change by deleting four test valves It was determined that the test valves were not necessary and only increased the maintenance costs and the exposure of personnel.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that for revision 0 and revision 1 of this design change, the modified system meets the Seismic Class I criteria.

9.1.4 Radiation Monitoring System Upgrade

Description of Change:

RFC-DC-12-3076 revision 1 allowed for the replacement of the Westinghouse incore instrument room radiation monitor inside containment with an Eberline radiation monitor which has no safety function. This new radiation monitor now is connected to an existing balance of plant Eberline data acquisition monitor located outside of the containment.

Safety Evaluation Summary:

This change, Unit 1 subtask 4, was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that these monitors are meeting the intent of the radiation monitoring system to prevent unmonitored and uncontrolled releases by using functionally equivalent Eberline equipment.

9.1.5 Pipe Support Modifications

Description of Change:

RFC-DC-12-3080 was written as a general design change to cover the work involved in the performance of pipe support modifications as discrepancies between the as-found support configuration and the originally intended design were identified. Modifications performed under RFC-DC-12-3080 result in a pipe and pipe support configuration that are in conformance with the UFSAR design basis requirements.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the modifications made to the piping supports would only be made to restore the piping supports (and hence the associated systems) to their UFSAR design. The design and installation of the piping supports would be in accordance with existing approved design change procedures, and any required pipe support or pipe stress analysis will be performed using existing approved analytical methods.

9.1.6 Large Bore Piping Reconstitution Program (LBPRP)

Description of Change:

RFC-DC-12-3081 was written as a general design change to cover the work involved in the performance of pipe support modifications as discrepancies between the as-found support configuration and the originally intended design are identified by the LBPRP. Modifications performed under RFC-DC-12-3081 result in a pipe support configuration that ensures the pipe support and associated piping system are in conformance with UFSAR design basis requirements.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the modifications made to the piping supports would only be made to restore the piping supports (and hence the associated systems) to their UFSAR design. The design and installation of the piping supports would be in accordance with existing approved design change procedures and any required pipe support or pipe stress analysis will be performed using existing approved analytical methods.

9.1.7 Reactor Vessel Head Vent Piping Modification

Description of Change:

RFC-DC-12-3092 revision 0 allowed for the modification of the piping associated with the reactor vessel head vent, because Westinghouse had identified an operational scenario which was not considered when the system was designed (single pathway operation) and because the effect of the thermal hydraulic load due to rapid valve opening was not considered in the original design. The modifications are to add a thermal expansion loop and modify supports. Revision 1 allowed for the change of the orifice size in the reactor vessel head vent piping to limit the flow so that a break of the line downstream of the orifice will not create a small break LOCA.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that, for revision 0, this modification will lower the stresses in the piping when only one valve is opened, and the modified system will meet its design requirements for a seismic class I system. Also, for revision 1 this modification results in an orifice that will still limit flow to less than the makeup capability.

9.1.8 Replacement of Instrument Port Conoseal Assemblies

Description of Change:

RFC-DC-12-3120 allowed for the replacement of the existing reactor vessel instrument port conoseal assemblies with ABB - Combustion Engineering canopy welded core exit thermocouple assemblies (CETNA). The new seal assembly will allow easier and faster reactor disassembly and assembly during refueling and is intended to reduce failure and leakage.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the new seal assembly will replace the existing assembly using existing ASME Boiler Code and seismic criteria.

9.1.9 Relocate Unit 2 Turbine Driven Auxiliary Feedwater Pump Flow Orifice

Description of Change:

RFC-DC-02-4126 revision 0 allowed for the relocation of the unit 2 turbine driven auxiliary feedwater pump discharge flow orifice to a location further downstream of the existing location but still within the pump room. This change is to correct inaccuracies associated with the orifice. Revision 1 also removed an instrument rack, relocated some instruments and changed a flow transmitter.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that design, procurement and installation will be performed in accordance with appropriate procedures, standards, codes and guidelines. This design change fulfills a commitment made to the NRC to correct the inaccuracy associated with the orifice (reference: LER89-017, revision 4).

9.2 PLANT MODIFICATIONS (PMs)

9.2.1 Personnel Exit in Fire Zone (FZ) 89

Description of Change:

02-PM-836 allowed for the installation of a second personnel exit in FZ 89, the turbine room Unit 2 miscellaneous oil room. Per AEP:NRC:0692BY, we are committed to satisfy National Fire Protection Association (NFPA) Code, Section 30. NFPA Code, Section 30, references NFPA Code, Section 101, for the design of exit facilities. This design change was done to meet the NFPA Code, Section 101. The door enhances personnel safety and the only impact to the UFSAR is that Figure 1.3-7 shows only one door for FZ 89.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that this PM adds a second personnel exit to FZ 89, it is not safety related, and has no effect on the UFSAR text or UFSAR accidents.

9.2.2 Meteorological System Instrumentation

Description of Change:

12-PM-1352 allowed for the removal of four strip chart recorders and removes two dew point sensors to improve the reliability of the meteorological system since these components have been chronic maintenance items. Digital means are currently used to collect data from the chart recorders while the data associated with the dew point sensors are not used for any atmospheric stability calculations.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that this PM does not affect any equipment needed to mitigate an accident evaluated in the UFSAR.

9.2.3 Installation of Electrical Distribution Panels

Description of Change:

12-PM-830 allowed for the provision of accessible power supplies for maintenance and construction activities during outages on either unit. The non-safety 480 V panels are located on the main turbine floor and near the refueling water storage tank for trailers located outside. In the past, power cables for these activities were run across open areas, such as the 633' main turbine floor, and thus were an personnel hazard. Also, during outages a personnel had to wait for available power lines due to electrical overload concerns.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the distribution panels: are fed from 600V BOP buses, are used only during outages, are not necessary for accident mitigation and feed only non-safety related loads. Also, postulated interference concerns with components important to safety were resolved.

9.3 MINOR MODIFICATIONS (MMs)

9.3.1 Replacement of Reactor Cavity Seal System

Description of Change:

12-MM-556 allowed for the replacement of the previous reactor cavity seal system, which is not safety-related, with a mechanical seal system. This eliminates the need to purchase a new reactor cavity seal for each refueling outage and also eliminates the need to use room temperature vulcanizing rubber to enhance the sealing characteristics of the seal. The new seal uses a lever system to exert pressure on the seal which is on top of the horizontal surfaces of the reactor vessel cavity floor and the reactor vessel flange. The previous inflatable "Pressray" seal was a vertical seal between the reactor vessel cavity floor and the reactor vessel flange.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the new mechanical seal design is superior to the existing seal design thus there was no degradation of seal function.

9.3.2 Modify the Lower Shelf Plate in Containment Penetration (CPN-71)

Description of Change:

12-MM-510 allowed for the shortening of a shelf plate inside CPN-71 by approximately 18 inches so that, when the containment penetration isolation cover is installed during a refueling outage, it does not interfere with the lower shelf plate.

The shelf plates are horizontal and help to support the hoses and cables passing through the penetration. During normal operation there are two hinged flange covers installed on the penetration.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the seismic qualification of the penetration was not degraded and the containment boundary function of the penetration was not impacted.

9.3.3 Installation of Control Rod Drive Mechanism (CRDM) Canopy Seal Clamps

Description of Change:

12-MM-534 allowed for the installation of mechanical seal clamps for leaking CRDM lower canopy seal welds. It also replaced with Belleville washers, washers used on the existing seal clamp. This same mechanical seal clamp design was used previously to repair a spare CRDM lower canopy seal weld leak.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the seismic class I design will be maintained and there will be no adverse impact on safety related equipment or seismic class I structures.

9.3.4 Redirect Discharge Lines for Safety Valves

Description of Change:

MM 02-MM-215 allowed for the modification of the outlet piping of component cooling water (CCW) safety valves 2-SV-72E and 2-SV-72W such that the safety valves discharge to the floor drain system rather than back to the CCW system. The valves are small sentinel valves located on the CCW (shell) side of the residual heat removal (RHR) heat exchangers.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the piping will be maintained seismic class I up to and including the valve, and the drain capacity is adequate to preclude flooding concerns. The routing of the safety valve discharge to a floor drain is intended to decrease malfunctions associated with the valve. The backpressure on the valves has in the past resulted in cap leakage and valve damage.

9.4 TEMPORARY MODIFICATIONS (TMs)

9.4.1 Fire Protection

Description of Change:

TM-2-94-03 and TM-2-94-20 allowed for hoses to be attached to drain valves 2-FHC-47-VI and 2-FHC-48-VI and connected them to the pre-piped sprinkler systems in the two contractor trailers in the turbine building. There are three sprinkler heads in each of the two contractor trailers to provide for fire suppression.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the amount of water to be used by the sprinklers is not significant with respect to the Fire Suppression system and is consistent with the defense in-depth fire protection program so that the plant fire protection is enhanced by this TM.

9.4.2 Unit 1 Instrumentation Replacement

Description of Change:

TM-01-94-09, -11, -12, -13, -14, -19, -21, -22, and -23 are TMs that were prepared in support of the instrumentation (H-line) replacement design change DC-12-RFC-2985. TM-01-94-09, -11, -12, -14, -21, -22, allowed for the installation of temporary power supplies for various instruments. TM-01-94-13, -19, and -23 allowed for the installation of jumpers to maintain the availability of various instruments.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that: these TMs were installed after Unit 1 had achieved mode 5, instrument indication was maintained, and there was continued supplementary indication of those parameters important for supporting reactivity control and decay heat removal; administrative and procedural controls were put in place to compensate for the disabled automatic functions.

9.4.3 Unit 2 Instrumentation Replacement

Description of Change:

TM-02-94-21, -22, -23, -24, -25, -26, -27, -28, -29, and -30 are TMs that were prepared in support of the instrumentation (H-Line) replacement design change DC-12-RFC-2985. These TMs allowed for the installation of temporary power supplies and/or installed jumpers for the instrumentation.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that: this TM was installed after mode 5, instrument indication was maintained, there was continued supplementary indication of those parameters important to supporting reactivity control and decay heat removal, and administrative and procedural controls were in place to compensate for the disabled automatic functions.

9.4.4 Refueling Water Storage Tank (RWST) Header

Description of Change:

TM-02-94-31 allowed for the installation of a plug in the 24 inch header from the RWST. This plug allowed work on various valves in the emergency core cooling system. This plug was used only during the period of time when the ECCS and containment spray systems are not required to be operable.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the core is completely unloaded and stored in the spent fuel pool and so the ECCS and containment spray systems are not needed during this period. Also the corrosion impact of the Neoprene plug on the stainless steel piping was determined to be insignificant during the short installation period.

9.4.5 Repair of Body-to-Bonnet Leak

Description of Change:

TM-02-94-35 allowed for the repair of the body to bonnet leak in motor operated valve 2-IMO-128 in the Loop 2-RHR cooldown line. The repair consists of installing a clamp around the valve to contain injected Furmanite.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the seismic class I rating of the RHR piping and the valve will be maintained, and that Furmanite will not adversely impact any safety systems.

9.4.6 Repair of Flange Leak

Description of Change:

TM-02-94-38 allowed for the repair of a leak associated with the flange gasket of check valve 2-FW-132-4, in the line from the west motor driven auxiliary feedwater pump to the number four steam generator. The flange gasket was repaired by installing a housing around the valve to contain injected Furmanite.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the seismic class I rating of the auxiliary feedwater piping system and the valve will be maintained and the valve itself as well as other safety related equipment in the area will not be adversely affected.

9.4.7 Circulating Water System Treatment

Description of Change:

TM-1-94-40 allowed for the installation of a vendor supplied, temporary sodium hypochlorite system to support zebra mussel and slime control in the circulating water system (specifically the main and feed pump condensers).

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that the installation of a temporary sodium hypochlorite chemical feed system has no impact on the accidents previously evaluated in the UFSAR and does not adversely impact the safety related essential service water system.

9.4.8 Winterization of The East Main Steam Enclosure

Description of Change:

TM-1-94-50 and TM-2-94-36 allowed for eight drain holes located in the Unit 1 and Unit 2 east main steam enclosures to be covered with Herculite. These holes were originally installed to provide a drainage path for water which would be released following a postulated break of a feedwater line. During the winter months, those holes allowed outside air to enter the enclosure creating the potential for freezing of the instrument lines.

Safety Evaluation Summary:

This change was reviewed and determined that it did not constitute an unreviewed safety question. This conclusion is based on the fact that existing pathways between various plant sections provide sufficient distribution of the water released by the accident to preclude the flooding of any required equipment.