TECHNICAL EVALUATION REPORT

AUDIT OF SUSQUEHANNA UNIT 2 TECHNICAL SPECIFICATIONS

PENNSYLVANIA POWER & LIGHT COMPANY
SUSQUEHANNA STEAM ELECTRIC STATION UNIT 2

NRC DOCKET NO. 50-388

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 25

FRC TASK 524

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

Mr. Joseph A. Murphy contributed to the technical preparation of this report through a subcontract with Schneider Consulting Engineers.

Mr. L. Briggs, NRC Region I Inspector, accompanied the FRC personnel during performance of the audit.

1. INTRODUCTION

1.1 PURPOSE OF AUDIT

The objective of the audit was to assist the Nuclear Regulatory Commission (NRC) in determining whether the selected plant technical specifications are compatible with the as-built safety-related systems, structures, and components of Susquehanna Unit 2. This technical evaluation report documents the results of that audit.

1.2 GENERIC BACKGROUND

During the low-power testing phases at Grand Gulf Unit 1, it was found that discrepancies existed between the technical specifications and the final safety analysis report (FSAR), the NRC safety evaluation report (SER), and the plant's as-built condition. Many of these discrepancies have been eliminated by amendments to the low-power license and by changing the technical specifications. In order to gain additional assurance that the Grand Gulf technical specifications were in agreement with the safety evaluations and the as-built condition, comparative audits were performed.

As a result of the problems found at the Grand Gulf plant, the NRC decided to conduct similar audits at the LaSalle plant, Washington Nuclear Plant 2, and Susquehanna Unit 2 to provide assurance that the plant technical specifications are compatible with the as-built plant.

1.3 PLANT-SPECIFIC BACKGROUND

On March 6, 1984, Franklin Research Center (FRC) was requested to assist NRC in performing an audit at Susquehanna Unit 2 to ensure that the plant technical specifications for selected safety-related systems are compatible with the as-built safety-related systems, structures, and components of the plant. The audit was to establish that hardware, its operating characteristics, and/or other conditions of the as-built safety-related systems, structures, and components are compatible with the parameters, descriptions, or

other information set forth in the selected technical specifications. The following scope of work for the audit was developed by NRC and discussed with the FRC auditors at the Region I office on March 7, 1984:

1. Primary containment isolation system (PCIS) valves: A sample of 25 PCIS valves were to be selected. The as-built condition of the valves and the surveillance procedures for the PCIS valves were to be reviewed to provide assurance that the as-built condition reflected the plant technical specification descriptions.

2. Drywell to suppression chamber vacuum breakers: Two vacuum breakers were to be selected at random. The as-built condition of the vacuum breakers was to be verified to be in accordance with the plant technical specification descriptions, and the existence of adequate surveillance procedures to address plant technical specification testing requirements was to be verified.

3. Automatic depressurization system (ADS) valves: The as-built condition of the ADS valves and associated surveillance procedures were to be reviewed to assure that plant technical specification requirements were adequately addressed.

4. Secondary containment ventilation system automatic isolation dampers: A physical verification was to be performed to determine if secondary containment ventilation system supply dampers were required to be addressed in Technical Specification Table 3.6.5.2-1.

5. Suppression pool volume: Because a discrepancy existed between the technical specification maximum water level and the FSAR maximum level for the suppression pool, the audit team was to identify the reason for the discrepancy.

6. Diesel generator day tank: The audit team was to examine an installed day tank to verify that sufficient volume existed to comply with plant technical specification requirements and that adequate surveillance procedures existed to provide assurance that the technical specification volume could be maintained.

This scope considered the following:

1. EG&G comparison of technical specifications with the FSAR; findings discussed by telephone with NRR representatives

2. NRC Region II Inspection Report 50-416/84-06 for the Grand Gulf plant; findings discussed with responsible Section Chief

3. previous Susquehanna problems (e.g., PCIS valves)

4. dominant BWR plant risk contributors; Susquehanna probabilistic risk analysis results (preliminary)

5. those technical specifications which will be verified during start-up program inspections.

2. EVALUATION

This section presents an item-by-item evaluation of compatibility of the plant technical specifications with the as-built condition of the plant for the primary containment isolation system (PCIS) valves, drywell to suppression pool vacuum breakers, automatic depressurization system (ADS) valves, secondary containment isolation dampers, suppression pool volume, and diesel generator day tank level indication.

2.1 PRIMARY CONTAINMENT ISOLATION VALVES

2.1.1 Scope

The task required review of the technical specifications, plant drawings, and the as-built condition of 25 PCIS valves to assure that:

a. the as-built condition reflects the description contained in the technical specifications

b. the technical specification testing requirements are adequately addressed by surveillance procedures

c. the electrical schematic drawings indicate that the isolation signals noted on Table 3.6.3-3 of the technical specifications are applied to actuate the valves.

2.1.2 Discussion

Twenty-seven PCIS valves were physically inspected. Appendix A contains a list of these valves and the nameplate data recorded.

The following documents supplied by the Licensee were reviewed:

- Piping and instrumentation drawings (P&IDs) for the following systems:
  - residual heat removal
  - reactor water clean-up
  - high pressure coolant injection
  - nuclear boiler - main steam
  - reactor recirculation

containment atmosphere control

reactor core isolation cooling

reactor building chilled water

reactor building component cooling water.

- Surveillance Procedure SO-259-011, "18 Month Manual Initiation of Drywell Cooling Automatic Isolation System".
- Surveillance Procedure SI-283-523, "18 Month Logic System Functional Test of Main Steam Line Isolation-Closure, Half Scram Channels A1, A2, B1, and B2"
- Surveillance Procedure SI-283-501, "Main Steam Line Isolation Logic System Functional Test"
- Surveillance Procedure SO-249-005, "Residual Heat Removal (RHR) Division I and II Quarterly Valve Exercising."

In addition, the data obtained by the Licensee from the performance of Surveillance Procedure SO-249-005 were reviewed for compliance with plant technical specifications.

2.1.3 Observations

- The reviewed surveillance procedures are in agreement with the plant technical specification requirements.
- The valve stroke times for 7 of the 27 valves reviewed were verified from the data recorded in Surveillance Procedure SO-249-005 and were found to be within the plant technical specification limits.
- No discrepancies were identified for the 27 PCIS valves reviewed.

2.1.4 Discrepancies

None.

2.1.5 Recommendations

During review of the P&IDs by the auditors, several valves were determined to be first isolation valves outside the primary containment penetrations, but they were not listed as containment isolation valves. Discussions with the Licensee revealed that exemptions for these valves had been requested

in the PSAR [1] and were granted in the plant SER [2, p. 6-33]. Review of the plant surveillance procedures indicated that these valves were subject to essentially the same testing requirements as valves listed as PCIS valves in the technical specifications with the exception of local leak rate testing. Local leak rate testing could not be performed because no isolation valve exists inside primary containment. With these valves excluded from the technical specification PCIS valve listing, there is concern that the surveillance requirements could be significantly modified or eliminated by the Licensee. To obviate this concern, it is recommended that the valves be added to the technical specification PCIS valve table with the leak rate testing exception noted.

2.2 DRYWELL TO SUPPRESSION POOL VACUUM BREAKERS

2.2.1 Scope

The task required selection of two vacuum breakers to assure that they are tested to meet technical specification requirements and that their installation agrees with the technical specification description.

2.2.2 Discussion

It was not possible to physically inspect the vacuum breakers installed between the drywell area and the suppression pool because the wetwell air space had been inerted and current atmospheric samples (oxygen, nitrogen, and other gases) were not available to allow authorization for entry. Documentation for the vacuum breakers was reviewed to assure that the actions required for proper vacuum breaker operation are being performed by the Licensee.

The following documents supplied by the Licensee were reviewed:

- FSAR Section 6.2.1.1.3.2
- Surveillance Procedure SO-259-002, "Operability Check of Suppression Chamber Drywell Vacuum Relief Breaker Valves"
- Surveillance Procedure SM-259-002, "18 Month Vacuum Relief Breaker Valve Set Pressure Test"
- Technical Manual for Suppression Pool Drywell Vacuum Relief Breaker Valves - 10M 166

- Surveillance Procedure for Containment Exit
- Technical Specifications for the Suppression Chamber Drywell Vacuum Breaker Relief Valves.

In addition, Surveillance Procedure SO-259-002 had been performed on February 14, 1984, and the results of the completed procedure were reviewed for compliance with plant technical specification requirements.

2.2.3 Observations

- The set pressure and set pressure tolerance for the vacuum breakers required by the technical specifications are verified by surveillance procedure SM-259-002 at 18-month intervals. The vendor technical manual (10M 166) data are also in agreement with technical specification requirements and Surveillance Procedures SM-259-002 and SO-259-002 requirements.
- Operability testing performed under Surveillance Procedure SO-259-002 is in accordance with plant technical specifications and the vendor technical manual instructions.
- Verification that the covers for the vacuum breakers are in the proper position is established in the surveillance requirements contained in the containment exit procedure and is in accordance with technical specification requirements.

2.2.4 Discrepancies

- The technical specifications do not contain any specific requirements for setting and calibration of the limit switches for the vacuum breakers.
- The vendor technical manual indicates that the limit switches should indicate that the valve is fully closed or fully open.
- The surveillance procedures require verification that the limit switches are operable and properly calibrated. However, the procedures do not contain any information on how to calibrate the limit switches or on what are considered acceptable data.
- The FSAR, on page 6.2-5, states:

  "Each of the inboard vacuum breakers is connected to a common alarm which indicates when any valve is not fully closed. Each of the outboard vacuum breakers is connected to a common alarm which indicates when any valve is not fully closed. There is individual

vacuum breaker position indication in the main control room for each valve.

The normally closed switches are held open when the valve is fully closed. The switches have a hysteresis or differential travel of 0.025". The switch hysteresis is multiplied through the mechanical linkage so that when the valve is opening under differential pressure the disk of the inboard valve is 0.32" off the seat before the "not fully closed" light comes on. The outboard valve can be 0.2" off the seat under similar conditions. When the valve is closing under differential pressure or when the valve is opening or closing by the actuator, the mechanical linkage assures that the "not fully closed" light is on unless the disk is on the seat."

There are no similar requirements in the plant technical specifications. The discrepancy was brought to the attention of the Licensee and the NRC resident inspectors.

2.2.5 Recommendations

The Licensee should determine the reason for the discrepancy between the technical specifications and the FSAR regarding limit switch setting and alibration for the suppression chamber drywell vacuum relief breaker valves. The surveillance procedures should be revised as required to ensure that the limit switches are properly calibrated in accordance with the FSAR description.

2.3 AUTOMATIC DEPRESSURIZATION SYSTEM VALVES

2.3.1 Scope

The task required verification that the installed condition of the ADS valves and the plant surveillance procedures reflect the plant technical specification requirements.

2.3.2 Discussion

Two of the ADS valves were physically inspected for proper installation; nameplate data were recorded and are included in Appendix B.

The following documents supplied by the Licensee were reviewed:

- FSAR Section 6.3.2.2.2
- Plant Technical Specifications Section 3.4.5.1, Table 3.3.1-1

- Surveillance Procedures SO-283-001, SO-283-002, SI-280-303, SI-283-321, and SI-283-322.

2.3.3 Observations

- The installed condition of the valves is represented by the plant drawings.
- The reviewed surveillance procedures establish testing requirements in accordance with the plant technical specification requirements.

2.3.4 Discrepancies

None.

2.3.5 Recommendations

None.

2.4 SECONDARY CONTAINMENT ISOLATION DAMPERS

2.4.1 Scope

The task was to assure that Technical Specification Table 3.6.5.2-1 contains supply dampers for the secondary containment ventilation if necessary. In addition, discussions at Region I indicated that the standby gas treatment systems (SGTS) dampers required specific review.

2.4.2 Discussion

Technical Specification Table 3.6.5.2-1 identifies those valves and dampers that are part of the secondary containment (reactor building) boundary. No dampers associated with the SGTS are listed on the referenced table. Discussions with the Licensee and review of the P&IDs revealed that the SGTS dampers open on a secondary containment isolation and are not used for isolation purposes.

A subsequent review of the normal HVAC supply and exhaust dampers revealed that many of the secondary containment isolation dampers were not

included in the draft Technical Specification Table 3.6.5.2-1 for Unit 2 or the issued Unit 1 Technical Specifications. However, discussions with the NRC licensing Project Manager for Susquehanna Unit 2 revealed that a revised Table 3.6.5.2-1 had recently been issued in NUREG-1042 (Susquehanna Unit 2 Technical Specifications). A copy of this table was provided to the auditors for review. The revised Table 3.6.5.2-1 addresses all secondary containment ventilation system isolation dampers identified on the flow diagrams.

2.4.3 Observations

- The revised Table 3.6.5.2-1 contained in NUREG-1042 adequately addresses all secondary ventilation system isolation dampers.
- The SGTS dampers do not provide any secondary containment isolation function; isolation is provided by the normal HVAC dampers.

2.4.4 Discrepancies

None.

2.4.5 Recommendations

With the present configuration of the SGTS, the supply dampers (i.e., crossover) between Zone I (Unit 1 reactor building) and the recirculation plenum are not required to provide a secondary containment isolation function in the event of a Unit 2 secondary containment ventilation system isolation. Should additional modifications be made to the SGTS to return the system to its original design (two-zone operation on a secondary containment ventilation system isolation), the supply dampers to the recirculation system plenum for Zone I (Unit 1 reactor building) should be added to the Unit 2 Technical Specifications as secondary containment ventilation system isolation dampers; similarly, the Zone II supply dampers to the recirculation system plenum should be included as secondary containment ventilation system isolation dampers to the Unit 1 Technical Specifications.

2.5 SUPPRESSION POOL VOLUME

2.5.1 Scope

The task was to identify the reason for a discrepancy in suppression chamber water volume between the technical specifications and the FSAR.

2.5.2 Discussion

Table 6.2.1 in the FSAR lists the following information concerning the suppression chamber water volume.

Minimum, ft^3	122,410
Maximum, ft^3	131,550
Pool depth (normal), ft	23

Section 3/4.6.2 of the technical specifications, "Depressurization Systems," states the following:

"Suppression Chamber ‡

Limiting Condition for Operation

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 133,540 ft^3 and 122,410 ft^3, equivalent to a level between 24'0" and 22'0"."

The minimum volumes identified in the technical specifi ions and the FSAR are in agreement. The normal pool depth maintained by the Licensee is 23 ft and is in agreement with the FSAR and the technical specifications.

2.5.3 Observations

- The Licensee stated that the suppression pool high-level alarm is set at 23 ft 9 in. (This level corresponds to a suppression pool volume of 132,147 ft^3.) The Licensee also stated that when the worst-case instrument error is considered, the high-level alarm point (23 ft 9 in) ensures that the maximum pool level of 24 ft, corresponding to a volume of 133,540 ft^3, is not exceeded.

2.5.4 Discrepancies

The information available from the Licensee does not provide a technical explanation for the discrepancy in the maximum suppression pool water volume identified in the FSAR and the technical specifications. (The technical specification value is 1990 ft^3 greater than the FSAR.)

2.5.5 Recommendations

The Licensee should determine the reason for the discrepancy in maximum suppression pool water volume and whether this discrepancy has any effect on suppression chamber performance. If the value in the FSAR is correct and the value in the technical specifications requires revision to conform to the FSAR, the high-level alarm setpoint will require readjustment to a lower value. Further, if this instrument error argument is valid, the low-level alarm setpoint should also be reevaluated.

2.6 DIESEL GENERATOR DAY TANK LEVELS

2.6.1 Scope

The scope of work included verification of the diesel generator day tank volume and surveillance requirements to assure that a minimum acceptable level is maintained in the day tank.

2.6.2 Discussion

The "A" diesel generator day tank was inspected to ensure that a method for verification of day tank level is provided and surveillance procedures are established.

2.6.2 Observations

- The diesel generator day tank is provided with level instrumentation to indicate the tank level and alarm at the low-level setpoint. Plant surveillance procedures require verification of the tank level in accordance with technical specification requirements.

2.6.3 Discrepancies

None.

2.6.4 Recommendations

None.

2.7 OTHER OBSERVATIONS

During the physical inspection of the PCIS valves, several observations concerning the as-installed condition of the valves were made. They are as follows:

- o Namco limit switches: Several Namco limit switches located inside primary containment were identified as models environmentally qualified for outside primary containment use only. Discussions with the Licensee revealed that replacement of these limit switches was in progress. In addition, Namco limit switches for use inside containment were environmentally qualified with the electrical connection provided with a sealant. The limit switches observed at the plant used a standard strain-relief connection with no sealant evident.

- o Junction boxes: The junction boxes inspected inside primary containment did not have pressure equalization (weep) holes necessary to ensure post-accident environmental qualification for high pressure.

3. CONCLUSIONS

The audit confirmed that a discrepancy exists between the maximum allowable suppression chamber volume in the FSAR and plant technical specifications. The audit also revealed a lack of quantified acceptance criteria in calibration procedures for the suppression chamber to drywell vacuum breakers. No further discrepancies were identified among the FSAR, the technical specifications, and the as-built conditions for the equipment evaluated.

4. REFERENCES

1. Susquehanna Steam Electric Station Units 1 and 2
   Final Safety Analysis Report, Pennsylvania Power and Light Company
   September 1978

2. NUREG-0776, including Supplements 1 through 5, "Safety Evaluation Report related to the operation of Susquehanna Steam Electric Station, Units 1 and 2," USNRC, April 1981

3. NUREG-0831, "Technical Specifications, Susquehanna Steam Electric Station, Unit 2," USNRC, September 1983

APPENDIX A

PRIMARY CONTAINMENT ISOLATION VALVES REVIEWED

The information contained within this appendix was compiled from the controlled as-built piping and instrumentation diagrams at the Susquehanna Unit 2 site and from the nameplates on the primary containment isolation valves. Where information is noted as "not accessible," physical obstructions (insulation, etc.) prevented the recording of data.

Valve Number:	HV-241F028B
Function:	Main Steam Isolation Valve
Type:	Atwood & Morril 26-in globe valve with electrohydraulic actuator
Location:	Penetration X-7B inboard; Line No. --
Purchase Order:	8856M1; Specification No. 21A9257, Rev. 3
Serial Number:	SN11221
Accessories:	Solenoid valves, Gould Allied Control Serial No. SV24123C, C1, C2 Limit switch, Namco, EA740, EA700

Valve Number:	HV-241F022C
Function:	Main Steam Isolation Valve
Type:	Atwood & Morril 26-in globe valve with electrohydraulic actuator
Location:	Penetration X-7C inboard; Line No. 26-G001
Purchase Order:	8656M1, Specification No. 21A9257, Rev. 3
Serial Number:	Not accessible
Accessories:	Solenoid valves, Gould Allied Control Limit switches, Namco, EA740, EA700

Valve Number:	HV-241F016
Function:	Main Steam Line Drain Valve
Type:	3-in gate valve with Limitorque motor operator
Location:	Penetration X-83 inboard; Line No. 3-DBA-208
Purchase Order:	Not accessible
Serial Number:	Not accessible
Accessories:	Limitorque motor operator, size SMB-00 Serial No. 21657 Reliance electric motor, Class RH insulation

Valve Number: HV-255F002
Function: HPCI Steam Supply
Type: 10-in gate valve
Location: Penetration X-11 inboard; Line No. 10-DBA-202
Purchase Order: Not accessible
Serial Number: Not accessible
Accessories: Limitorque motor operator, Size SMB-1
Serial No. 218058
Reliance electric motor, Class RH insulation
Namco EA170 limit switches (2)

Valve Number: HV-255F100
Function: HPCI Steam Supply
Type: 8-in Masoneilan globe valve with pneumatic actuator
Location: Penetration X-11 inboard; Line No. 8-DBA-202
Purchase Order: 8856-J065BAC
Serial Number: N00186-5-2
Accessories: Air operator, Masoneilan Model No. 38-2076L-9-9
Solenoid valve, Asco NPKX8321A1E
Limit switches (2), Namco EA180

Valve Number: HV-255F006
Function: HPCI Injection
Type: 14-in Anchor Darling gate valve with Limitorque motor operator
Location: Outboard; Line No. 14-DBB-220
Purchase Order: 8856-P-10A
Serial Number: E-5853-49-1
Accessories: Limitorque motor operator, Size SMB-3
Serial No. 343670
Reliance electric motor, Class H insulation

Valve Number: HV-255F042
Function: HPCI Suction
Type: 4-in Anchor Darling with Limitorque motor operator
Location: Penetration X-94 outboard
Purchase Order: Not accessible
Serial Number: E-5853-70-1
Accessories: Limitorque motor operator, Size SMB-000
Reliance electric motor, Class B insulation

Valve Number: HV-251F009
Function: RHR Shutdown Cooling Suction
Type: 20-in globe valve with Limitorque motor operator
Location: Penetration X-12 inboard; Line No. 20-DCA-208
Purchase Order: 8856-P-17A
Serial Number: Not accessible
Accessories: Limitorque motor operator, Size SMB-1
Reliance electric motor, Class RH insulation

Valve Number: HV-241F104
Function: RWCU Return
Type: 4-in gate valve with Limitorque motor operator
Location: Penetration X-94 outboard
Purchase Order: Not accessible
Serial Number: Not accessible
Accessories: Limitorque motor operator
Reliance electric motor, Class B insulation

Valve Number: HV-244F004
Function: RWCU Suction
Type: 6-in gate valve with Limitorque motor operator
Location: Penetration X-14 outboard; Line No. 6-DBC-201
Purchase Order: Not accessible
Serial Number: Not accessible
Accessories: Limitorque motor operator, Size SMB-00
Serial No. 213611
Reliance electric motor, Class RH insulation
Limit switch, Namco EA170

Valve Number: HV-244F001
Function: RWCU Suction
Type: 6-in gate valve with Limitorque motor operator
Location: Penetration X-14 outboard; Line No. 6-DBC-201
Purchase Order: 8856-P-10A
Serial Number: Not accessible
Accessories: Limitorque motor operator, Size SMB-00
Serial No. 213453
Reliance electric motor, Class H insulation
Limit switch, Namco EA180

Valve Number: SV-22605
Function: Containment Instrument Gas
Type: 2-in Target Rock Model 75KK-285 solenoid operated globe valve
Location: Penetration X-80C outboard; Line No. 2-HCB-221
Purchase Order: 8856-J-70
Serial Number: SV12605
Accessories: None

Valve Number: SV-25752B
Function: Containment Atmosphere Sample
Type: 2-in Target Rock Model 75KK-211 solenoid operated globe valve
Location: Penetration X-80C outboard
Purchase Order: 8856-J-70
Serial Number: Not accessible
Accessories: None

Valve Number: HV-21345
Function: Reactor Building Component Cooling Water
Type: 4-in gate valve with Limitorque motor operator
Location: Penetration X-24 inboard; Line No. 4-HBD-230
Purchase Order: 8856-P-12-A
Serial Number: E9052-1-3
Accessories: Limitorque motor operator, Size SMB-00
Peerless electric motor, Class H insulation

Valve Number: HV-25713
Function: Containment Purge
Type: 24-in Henry Pratt butterfly valve with pneumatic actuator
Location: Penetration X-26 outboard; Line No. 24-HBB-217
Purchase Order: 8856-P-31-AC
Serial Number: D-0026-1-2
Accessories: Pneumatic operator, Bettis Model 1416-SR-3-M3
Solenoid valve, Circle Seal Controls Model SN-315-9101-3-B
Limit switch (2), Namco EA740

Valve Number: HV-243F019
Function: Reactor Coolant Sample
Type: 3/4-in Masoneilan globe valve with pneumatic operator
Location: Penetration X-60B inboard; Line No. 3/4-DCA-243
Purchase Order: 8856-J65-BAC
Serial Number: N00186-14-2
Accessories: Pneumatic operator, Masoneilan Model 38-200761-9-9
Solenoid valve, Asco NPKX8321A1E
Limit switch (2), Namco EA180

Valve Number: HV-243F020
Function: Reactor Coolant Sample
Type: 3/4-in Masoneilan globe valve with pneumatic operator
Location: Penetration X-60B outboard; Line No. 3/4-DCA-243
Purchase Order: 8856-J-69B-AC
Serial Number: N00186-15-2
Accessories: Pneumatic operator, Masoneilan Model 38-20
Solenoid valve, Circle Seal Controls Model SN-315-9101-1-B
Limit switch (2), Namco EA180

APPENDIX B

AUTOMATIC DEPRESSURIZATION SYSTEM VALVES

The following information was obtained from the nameplate data from two of the inboard automatic depressurization system valves. It should be noted that the valve tag numbers could not be located.

Valve 1: Crosby Electromatic Actuated Relief Valve
Direct Acting Safety Relief ASME III Class 1
Body and Bonnet ASME SA-105
Inlet Hydrostatic Pressure, 2370 psig
Outlet Hydrostatic Pressure, 975 psig
Drawing No. 08A63790
Solenoid Valve Serial Nos. S66274-279, S66274-273, S66274-272

Valve 2: Crosby Electromatic Actuated Relief Valve
Direct Acting Safety Relief ASME III Class 1
Body and Bonnet ASME SA-105
Inlet Hydrostatic Pressure, 2370 psig
Outlet Hydrostatic Pressure, 975 psig
Crosby Tag No. HV65BP-1N
GE Specification No. GE 22A6441
Solenoid Valve Serial Nos. S66274-289, S66274-305, S66274-285

EG&G Idaho, Inc.

P.O. BOX 1625, IDAHO FALLS, IDAHO 83415

March 27, 1984

Mr. F. L. Sims, Director
Reactor Research and Technology Division
Idaho Operations Office - DOE
Idaho Falls, ID 83401

TRANSMITTAL OF SUSQUEHANNA, UNIT 2, REPORT A6816 - LPL-109-84

Ref: J. M. Fehringer and J. C. Stachew, Audit of Nuclear Plant Technical Specifications Susquehanna Steam Electric Station, Unit 2, Docket No. 50-388, EGG-EA-6541, March 1984

Dear Mr. Sims:

Enclosed is the referenced final report. This report determined that there are inconsistencies between eight Technical Specification Sections, the Final Safety Analysis Report and the Safety Evaluation Report for Susquehanna Steam Electric Station, Unit 2. This report issued under FIN A6816 completes Node 106-D1 on the FY1984 NRC Support Milestone Chart.

Very truly yours,

J. C. Hunter for

L. P. Leach, Manager
Reactor Evaluation Programs

JMF:jh

Enclosure:
As Stated

cc: J. N. Donohew, NRC/DL (5)
G. C. Meyer, NRC/DL
J. O. Zane, EG&G Idaho (w/o Enc.)

EGG-EA-6541

March 1984

AUDIT OF NUCLEAR PLANT TECHNICAL SPECIFICATIONS

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

DOCKET NO. 50-388

J. M. Fehringer
J. C. Stachew

Idaho National Engineering Laboratory
Operated by the U.S. Department of Energy

This is an informal report intended for use as a preliminary or working document

~~8405290446 PDR~~

Prepared for the
U.S. NUCLEAR REGULATORY COMMISSION
Under DOE Contract No. DE-AC07-ID01570

EG&G Idaho

TABLE I. (Continued)

SECTION		CONSISTENT/INCONSISTENT
3/4.6.5	SECONDARY CONTAINMENT	
	Secondary Containment Automatic Isolation Dampers	Inconsistent
	Standby Gas Treatment System	Consistent
3/4.6.6	PRIMARY CONTAINMENT ATMOSPHERE CONTROL	
	Drywell and Suppression Chamber Hydrogen Recombiner Systems	Consistent
	Drywell and Air Flow Systems	Consistent
	Drywell and Suppression Chamber Oxygen Concentration	Consistent
3/4.8 ELECTRICAL POWER SYSTEMS		
3/4.8.1	A.C. SOURCES	
	A.C. Sources-Operating	Inconsistent
3/4.8.2	ONSITE POWER DISTRIBUTION SYSTEMS	
	Distribution - Operating	Inconsistent
	D.C. Sources - Operating	Inconsistent
	Primary Containment Penetration Conductor Overcurrent Protective Devices	Inconsistent

The FSAR does not identify specific designations for the 480VAC buses, the 125VDC and 250VDC fuse boxes. Therefore, the completeness of the T/S 3.8.3.1 cannot be verified.

8. T/S Section 3/4.8.4.1 (Primary Containment Penetration Conductor Overcurrent Protective Devices)

T/S Table 3.8.4.1-1 (Primary Containment Penetration Conductor Overcurrent Protective Devices) identifies the overcurrent protective devices required to determine electrical equipment operability.

The FSAR does not identify any of the overcurrent protective devices listed in T/S Table 3.8.4.1-1. Therefore, the completeness of T/S Table 3.8./4.1-1 cannot be verified.

Table I contains a summary of the Susquehanna-2 T/S sections reviewed; consistencies and inconsistencies with the FSAR and/or the SER are shown.

3. DISCUSSION

The following inconsistencies were identified:

1. T/S Section 3/4.3.2 (Isolation Actuation Instrumentation)

   The completness of T/S Table 3/4.3.3.2 (Isolation Actuation Instrumentation) cannot be verified by the FSAR Table 7.3-5 (Containment and Reactor Vessel Control System Instrumentation Specifications). A total listing/discussion of all instrument channels identified in T/S Table 3/4.3.3.2 are not addressed in FSAR Table 7.3-5.

2. T/S Section 3/4.6.2.1 (Suppression Chamber)

   The FSAR Section 6.2 page 6.2.1-92 identifies a maximum allowable water volume of 131,550 ft^3 in the suppression chamber. The T/S Limitng Conditions for Operation (LCO) 3.6.2.1 identifies a maximum allowable water volume of 133,540 ft^3 in the suppression chamber.

3. T/S Section 3/4.6.3 (Primary Containment Isolation Valves)

   T/S Table 3.6.3-1 (Primary Containment Isolation Valves), identifies isolation valve data (isolation timing and input signals) that cannot be matched with the isolation valve data in the FSAR Table 6.2-12 (Primary Containment Isolation Valve Summary). There is no correlation between the valve designations identified in the FSAR and in the T/S.

4. T/S Section 3/4.6.5.2 (Secondary Containment Automatic Isolation Dampers)

ABSTRACT

This report documents the review of the Susquehanna Steam Electric Station Unit 2 (Susquehanna-2) Technical Specifications (T/S) to determine if selected sections of the T/S are consistent with the Susquehanna Final Safety Analysis Report (FSAR) as amended and the Susquehanna Safety Evaluation Report (SER) as supplemented. Inconsistencies are listed in this report but no further evaluation was conducted to determine if the inconsistency was an indication of an error in any of the subject documents.

FOREWARD

This report is supplied as part of the "Audit of Nuclear Plant Technical Specifications" being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Licensing, by EG&G Idaho, Inc., NRC Licensing Support Section.

The U.S. Nuclear Regulatory Commission funded the work under authorization B&R 20 19 10 11 1 FIN No. A6816.

PP&L Pennsylvania Power & Light Company
Two North Ninth Street • Allentown, PA 18101 • 215 / 770-5151

MAR 2 6 1984

Norman W. Curtis
Vice President-Engineering & Construction-Nuclear
215/770-7501

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
CERTIFICATION OF UNIT 2 TECHNICAL SPECIFICATIONS
ER 100508 FILE 841-6
PLA-2114 Docket No. 50-388

Dear Mr. Denton:

In response to Mr. Eisenhut's letter dated March 5, 1984, attached are Pennsylvania Power & Light Company's proposed Unit 2 Technical Specifications. An amendment to our license application has been submitted which revises Section 16.2 of the Final Safety Analysis Report (FSAR) to reference this letter as containing our proposed Technical Specifications for Susquehanna SES Unit 2.

We have reviewed the draft Unit 2 Technical Specifications provided to us on February 17, 1984 including revisions received from the NRC staff on March 10 and 20, 1984. Based on that review, I certify that, to the best of my knowledge, the Technical Specifications for Susquehanna SES Unit 2 as proposed in this letter accurately reflect the plant, the FSAR and supplementary correspondence, and the SER analysis with the exception of the lack of an established limit on the measurement of secondary containment bypass leakage through the feedwater penetration.

The lack of this limit is not a safety concern since we are measuring the leakage and keeping the total leakage from this source and the MSL Drains to within 5.0 scfh, consistent with our analysis. A change to the existing Susquehanna SES Unit 1 Technical Specifications is in our internal review process. Prior to exceeding five percent power, we will submit a request to revise the Technical Specifications for both units to incorporate this limit.

The operating license for Susquehanna 1 was issued on July 17, 1982, and the startup of this unit, in our judgement, was very successful resulting in this unit being the first BWR since the TMI incident to achieve commercial operation. The experience with Unit 1 led to the identification of relatively minor clarifications of language that have been incorporated into the Unit 2 Technical Specifications. Additional changes from the Unit 1 document have been made to reflect the plant configuration of two units, to incorporate the effect of NRC staff resolution of industry generic issues, and to incorporate a number of minor administrative changes.

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Our overall assessment is that the Unit 1 Technical Specifications are sound and constitute a good basis for plant operation and monitoring compliance. The changes incorporated into the Unit 2 Technical Specifications, beyond those required to reflect system configuration, are relatively minor, but should contribute to elimination of misinterpretation in carrying out surveillances and operating this unit. Prior to exceeding 5% power on Unit 2, PP&L expects to request changes in the Unit 1 specifications to make them comparable to Unit 2.

If you have any comments or questions please contact us.

Very truly yours,

N W Curtis

N. W. Curtis
Vice President-Engineering & Construction-Nuclear

cc: R. L. Perch - NRC
R. H. Jacobs - NRC
D. R. Hoffman - NRC

DMB

MAR 20 1984

Docket No. 50-374

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Vice President
Post Office Box 767
Chicago, IL 60690

Gentlemen:

This refers to the special inspection conducted by Messrs. A. L. Madison, S. Stasek, S. Guthrie and D. Evans of this office on March 6 through 9, 1984, of activities at LaSalle County Station, Unit 2, authorized by NRC Operating License NPF-18, and to the discussion of our findings with Mr. R. D. Bishop at the conclusion of the inspection.

The enclosed copy of our inspection report identifies areas examined during the inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

No items of noncompliance with NRC requirements were identified during the course of this inspection.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosure(s) will be placed in the NRC Public Document Room unless you notify this office, by telephone, within ten days of the date of this letter and submit written application to withhold information contained therein within thirty days of the date of this letter. Such application must be consistent with the requirements of 2.790(b)(1). If we do not hear from you in this regard within the specified periods noted above, a copy of this letter and the enclosed inspection report will be placed in the Public Document Room.

IE01

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

"Original Signed [illegible] lius"

C. E. Norelius, Director
Division of Project and
Resident Programs

Enclosure: Inspection Report
No. 50-373/84-07(DPRP)

cc w/encl:
D. L. Farrar, Director
of Nuclear Licensing
G. J. Diederich, Station
Superintendent
R. H. Holyoak, Project Manager
DMB/Document Control Desk (RIDS)
Resident Inspector, RIII
Phyllis Dunton, Attorney
General's Office, Environmental
Control Division

RIII for | RIII | RIII
Walker/ld | Shafer | Norelius
03/16/84 | |

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-374/84-07(DPRP)

Docket No. 50-374 License No. NPF-18

Licensee: Commonwealth Edison Company
Post Office Box 767
Chicago, IL 60690

Facility Name: LaSalle County Nuclear Station, Unit 2

Inspection At: LaSalle Site, Marseilles, Illinois

Inspection Conducted: March 6 through 9, 1984

Inspectors: Roger D. Walker for
A. D. Madison
3-20-84
Date

Roger D. Walker for
S. Stasek
3-20-84
Date

Roger D. Walker for
S. Guthrie
3-20-84
Date

Roger D. Walker for
E. Evans
3-20-84
Date

Approved By: Roger D. Walker
R. D. Walker, Chief
Projects Section 2C
3-20-84
Date

Inspection Summary

Inspection on March 6-9, 1984 (Report No. 50-374/84-07(DPRP))
Areas Inspected: Special unannounced safety inspection to verify Technical Specification conformance to as-built plant configuration; review surveillance procedures and surveillance program implementation. The inspection involved a total of 80 inspector-hours by four inspectors.
Results: Of the two areas inspected, no items of noncompliance or deviations were identified.

DETAILS

1. Persons Contacted

   R. D. Bishop, Administrative and Support Services Assistant Superintendent
   J. C. Renwick, Technical Staff Supervisor

   The inspectors also talked with and interviewed various members of the Operations and Technical Staff.

2. Technical Specification (TS) Review

   At the request of the Office of Nuclear Reactor Regulation (NRR), Region III assigned four inspectors to review two sections of the Unit 2 Technical Specifications for technical adequacy and conformance to actual plant design:

   a. Section 3/4.6.3 Primary Containment Isolation Valves

   b. Section 3/4.8.2 Electrical Distribution

   The inspectors reviewed the applicable sections of the Final Safety Analysis Report (FSAR) to ensure Technical Specification conformance. The inspectors reviewed as-built drawings and performed in-plant walk-downs to verify that equipment in place matched that described in the Technical Specifications. The inspectors reviewed the Technical Specifications action statements for technical adequacy including verifying adequate electrical power for performance of all Emergency Core Cooling Systems (ECCS). The inspectors also reviewed the licensee's surveillance program to ensure compliance with Technical Specification requirements.

   (1) Section 3/4.6.3 Primary Containment Isolation Valves

   The inspectors' review of this section revealed two minor discrepancies:

   (a) Table 3.6.3-1 lists the primary containment isolation valves. However, under Automatic Isolation Valves, a. 14., only one Tip Guide Tube Valve Ball Valve; 2C51-J004 is listed. There are actually five valves; 2C51-J004A, B, C, D and E.

   The licensee's surveillance program and procedures recognize the existence of these five valves and the required testing and surveillance has been performed.

   (b) Table 6.2-21 in the FSAR requires a valve closure time of ≤ 40 sec for valves 2E12-F008 and 2E12-F009 (RHR shutdown cooling suction). However, T.S. Table 3.6.3-1 lists ≤ 41 secs for valve closure time and the licensee's surveillance procedures comply with the Technical Specifications.

The actual closure times as verified by recent testing is ≤ 35 sec for these valves.

Resolution and correction of these apparent discrepancies will be tracked as an unresolved item (374/84-07-01). No licensee action is required at this time.

During this review the inspectors also found several discrepancies in the licensee's surveillance matrix and procedures related to this Technical Specification section. These are discussed in Part 3 of this report.

(2) Section 3/4.8.2 Electrical Distribution

This section describes requirements for A.C. and D.C. electrical distribution for both operating and shutdown conditions. No discrepancies in the technical specifications were found.

However, the inspectors did find five deficiencies in the licensee's electrical drawings and labeling of breakers.

The licensee has committed to correcting these deficiencies and their action will be tracked as an open item (374/84-07-02).

Space A-2, MCC 235X-1 (2AP71E) and Space AA-4, MCC 235X-2 (2AP72E) appeared to have been additions to the motor control centers. Further review revealed no electrical loading concerns; however, the inspector questioned the effect these additions had on the seismic qualifications of the affected motor control centers. The licensee has committed to provide an analysis concerning the seismic qualification of those motor control centers performed by Sargent and Lundy, the architect-engineer. Resolution of this concern will be tracked as an open item (374/84-07-03).

3. Surveillance Matrix and Program

As part of the Technical Specification review, the inspectors reviewed portions of the licensee's surveillance matrix that were applicable to the sections of the Technical Specifications under review. The matrix is designed such that a direct written correlation between Technical Specification required surveillance, specific components and specific procedures should exist. The purpose of the review was to ensure that all components and surveillance for those components addressed in the Technical Specifications were listed in the matrix and that a procedure to perform that surveillance for each component existed. The inspectors also reviewed some procedures on a spot check basis to ensure that the procedure actually performed the required surveillance for the specific component and that the procedures were technically adequate.

a. Technical Specification 4.6.3.1 requires that each valve listed in Table 3.6.3-1 be tested to verify full travel and operability following maintenance. However, the matrix does not list an applicable procedure to fulfill this requirement for several valves.

Whether or not the required surveillance is actually being performed could not be determined without further information from the licensee. The licensee has agreed to provide the required information. Resolution of this matter will be tracked as an open item (374/84-07-04).

b. Valve 2E51-F069 was not listed in the matrix. However, further investigation confirmed that procedures existed to perform the required surveillance.

The matrix referred to LIS-NB-15 and 16 for the required surveillance on Excess Flow Check Valves, whereas the procedures were actually LIS-NB-115, 215, 116 and 216.

These and other minor discrepancies were noted and referred to the licensee. They will be corrected as part of an ongoing review by the surveillance group. This surveillance group was established February 1, 1982 and is charged with the responsibility of coordinating surveillance at LaSalle Station. The inspectors feel that this is a positive step and will enhance the licensee's performance in the surveillance area.

c. In the review of procedures it was noted that calibrated stopwatches were not required to perform closure time measurements. However, further investigation revealed that calibration stopwatches were actually being used. ANSI 18.7 (1976) Administrative Controls and Quality Assurance requires that procedures for tests and maintenance specify any special equipment to be used. The licensee has agreed to revise applicable surveillance procedures to require a calibrated stopwatch for measuring valve closure times. Completion of this will be tracked as an open item (374/84-07-05).

d. The inspectors also reviewed maintenance work requests to ensure that required surveillances were being performed following valve maintenance. No violations of requirements were identified. However, a potential source of confusion was identified in that the specific test requirement was not noted on the work request in all cases. Identifying the specific test requirements on the work request not only ensures that the desired tests are performed, but also allows Quality Control, Quality Assurance, and other reviewers the opportunity to verify that Technical Specification requirements are met. The licensee agreed that specific test requirements should be listed on the work request.

No items of noncompliance were identified.

4. Exit Interview

The inspector met with licensee representatives (denoted in Paragraph 1) at the conclusion of the inspection and summarized the scope and findings of the inspection activities. The licensee acknowledged those findings.

Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

March 21, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: LaSalle County Station Unit 2
Technical Specification Certification
NRC Docket No. 50-374

Reference (a): D. G. Eisenhut letter to Cordell Reed dated March 8, 1984.

(b): C.W. Schroeder letter to H. R. Denton dated January 13, 1984.

Dear Mr. Denton:

The purpose of this letter is to respond to Reference (a).

The interaction of LaSalle County Station personnel with the Standard Technical Specifications dates back to approximately 1974 when G. J. Diederich, then Assistant Superintendent for Operations, was a member of the BWR Standard Technical Specifications Committee. Mr. Diederich, who is now Station Superintendent at LaSalle County Station, thus gained first hand knowledge of the development philosophy, and NRC staff positions as they were incorporated into the original BWR Standard Technical Specifications (STS).

In 1978, Commonwealth Edison Company prepared the original draft of the LaSalle County Station Technical Specifications. This preparation included reviews by individual system test engineers, departmental reviews and a series of meetings with the entire operating staff to review the tech specs in detail. Further reviews were performed on a chapter by chapter basis by the NSSS vendor (GE) and the A/E (Sargent and Lundy). Following submittal of FSAR Chapter 16 (Amendment 39, October 1978), the NRC requested that future versions be submitted as marked up copies of the GE STS. This was performed as requested.

Two years prior to the Unit 1 license issue, the Tech Specs for Unit 1 were thoroughly reviewed by Commonwealth Edison Engineering, Station Staff, Nuclear Licensing, and Nuclear Safety for accuracy. These reviews included providing each page and all subsequent changes to the applicable system test engineers and other "experts" for review and comment. These comments were reviewed and many discussions were held within the Company and with NRR (Messrs. Bottimore, Bournia and reviewers).

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NRR issued many changes during this period (several dozen) to incorporate staff requirements, design changes and CECo requests. These changes also received multiple reviews by cognizant individuals.

During the almost two full years since the Unit 1 License NPF-11 was issued, it has been our experience that the Unit 1 Technical Specifications accurately reflect the plant and the FSAR. Certain specifications were found to have minor discrepancies that were either corrected by license amendments or were determined to be adequately controlled and identified in the Unit 1 Technical Specification upgrade to match Unit 2 (Reference b). The Unit 2 Tech Spec preparation started with the current Unit 1 Tech Spec at the time as the draft document and changes were made where differences existed. This was submitted to NRR as a draft. Additional changes were made and submitted in May, 1983 to D. Hoffman (NRR). These changes included improvements over Unit 1, clarifications, relaxations and new revised staff requirements where necessary. Such changes were held (at NRC request) for review and issuance at Unit 2 licensing, with the intention to then promptly backfit on Unit 1. The proof and review copy was received in August, 1983 and again was reviewed on site for accuracy by system test engineers and other "experts". Subsequently discussions were held with the staffs' reviewers including, a meeting at the Bethesda offices on September 20, 1983. Since the license condition identified in SSER Supplement 5 item 1.10(7)(1) on reactor containment electrical penetrations' redundant fault current devices was not issued due to installation of subject devices, a clarifying upgrade to the Unit 2 Technical Specification 3.8.3.2 will be submitted as an administrative change to note the backup devices.

The status of Unit 1 and Unit 2 Technical Specifications has been discussed on several occassions between the NRC Staff and Commonwealth Edison Company. During the Unit 2 operational readiness review meeting in Bethesda, Commonwealth Edison Company again stated our intention to upgrade the Unit 1 Technical Specifications to match the Unit 2 Technical Specifications. This action was agreed to by NRR management. On January 13, 1984, Reference (b) was submitted to fulfill our commitment. These changes were justified based on the fact that the NRC had just issued the exact same specifications on Unit 2 less than a month before (12/16/83). The NRC rejected this Unit 1 Technical Specification amendment request. Commonwealth Edison Company is in the process of reformatting our request and expect resubmittal in the near future.

It is our understanding that the NRC Region III recently concluded an extensive onsite review of the Technical Specifications for Containment Isolation and AC/DC power. We understand that review, which will be documented in an inspection report to be issued in the near future, concluded that those specifications are technically adequate.

Based upon the detailed, iterative process utilized to prepare the Unit 1 Technical Specifications, the positive two year operating experience with the Unit 1 Technical Specifications, the use of the Unit 1 Technical Specifications as the basis for the Unit 2 Technical Specifications, and the positive three month experience since the operating license was issued with the Unit 2 Technical Specifications, I conclude and certify that the Unit 2 Technical Specifications do accurately reflect the plant and the FSAR. Furthermore, I am satisfied that, because of these factors, no further adequacy reviews are warranted by Commonwealth Edison Company at this time.

Certain issues as to the interpretations of specifications and overly restrictive action statements that have been previously identified by Commonwealth Edison Company, owners groups, and NRR generic letters will continue to be pursued. Commonwealth Edison Company is also participating in the BWR Owners Group Technical Specification Improvements Committee and expects substantial changes in Technical Specifications to result from that effort. Finally, we are encouraged by the work that the NRC is initiating (NUREG-1024) to provide an overall upgrade of Technical Specifications.

To the best of my knowledge and belief the statements contained herein are true and correct. In some respects these statements are not based on my personal knowledge but upon information furnished by other Commonwealth Edison and contractor employees. Such information has been reviewed in accordance with Company practice and I believe it to be reliable.

Enclosed for your use are one signed original and thirty-nine (39) copies of this letter.

Very truly yours,

Cordell Reed

Cordell Reed
Vice President

lm

cc: Dr. A. Bournia - Telecopy
NRC Resident Inspector - LSCS

SUBSCRIBED and SWORN to
before me this 21st day
of March, 1984

Notary Public

8323N

EGG-EA-6539

March 1984

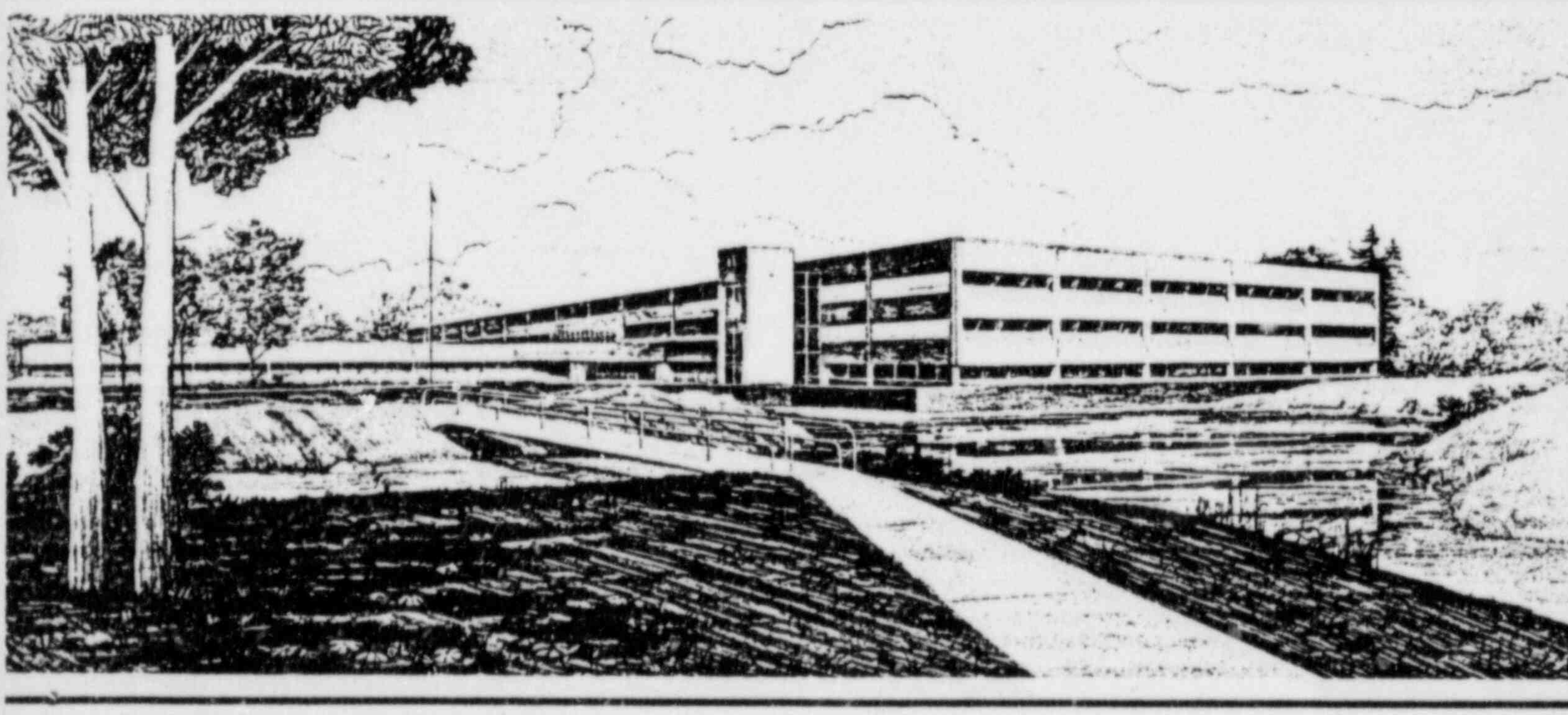
AUDIT OF NUCLEAR PLANT TECHNICAL SPECIFICATIONS

LA SALLE COUNTY STATION, UNIT 2

DOCKET NO. 50-374

J. M. Fehringer
D. M. Beahm

Idaho National Engineering Laboratory
Operated by the U.S. Department of Energy



This is an informal report intended for use as a preliminary or working document

~~8405110119 PDR~~

Prepared for the
U.S. NUCLEAR REGULATORY COMMISSION
Under DOE Contract No. DE-AC07-ID01570
FIN No. A6816

EG&G Idaho

EGG-EA-6539

LA SALLE COUNTY STATION, UNIT 2
AUDIT OF NUCLEAR PLANT TECHNICAL SPECIFICATIONS

Docket No. 50-374
TAC No. 54184

Published March 1984

J. M. Fehringer
D. M. Beahm

EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Responsible NRC Individual and Division:
C. Meyer/Division of Licensing

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6816

CONTENTS

TABLE

AUDIT OF NUCLEAR PLANT TECHNICAL SPECIFICATIONS

1. INTRODUCTION

The LaSalle County Station, Unit 2 (LaSalle-2) is a boiling water reactor (BWR) plant. It has been selected for an audit to determine if the LaSalle Technical Specifications (T/S)[1], are consistent with the LaSalle Final Safety Analysis Report (FSAR)[2] as amended, and the LaSalle Safety Evaluation Report (SER)[3] as supplemented. The specific sections of the T/S selected for audit and summary results are listed in Table I. Inconsistencies between these sections of the T/S and the FSAR and SER were identified but no further evaluation was conducted to determine if the inconsistencies were indications of error in any of the subject documents.

2. REVIEW CRITERIA

The T/S Limiting Conditions for Operation (LCOs) and Action Statements for each technical specification listed in Table I (Section 3) were compared with the FSAR and SER to determine if the T/S are consistent to the FSAR and SER. Emphasis was on the T/S Operational Mode 1, power operation, with exceptions noted in this report. Setpoints and lists of valves, instruments, overcurrent protective devices and electrical buses in the T/S were checked against tables in the FSAR and SER.

The SER was reviewed to ensure that requirements in the SER were addressed in the T/S.

The T/S bases and surveillance requirements were not reviewed in this audit of the T/S.

An explanation of each inconsistency between the T/S and the FSAR and SER is included in this report.

TABLE I. LASALLE-2 TECHNICAL SPECIFICATIONS/FSAR/SER CONSISTENCY SUMMARY

SECTION		CONSISTENT/INCONSISTENT
3/4.3 INSTRUMENTATION		
3/4.3.2	ISOLATION ACTUATION INSTRUMENTATION	Consistent
3/4.3.3	EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION	Consistent
3/4.5 EMERGENCY CORE COOLING SYSTEMS		
3/4.5.1	ECCS - OPERATING	Consistent
3/4.5.3	SUPPRESSION CHAMBER	Consistent
3/4.6 CONTAINMENT SYSTEMS		
3/4.6.1	PRIMARY CONTAINMENT	
	Primary Containment Integrity	Consistent
	Primary Containment Leakage	Consistent
	Primary Containment Air Locks	Consistent
	MSIV Leakage Control System	Consistent
	Primary Containment Structural Integrity	Consistent
	Drywell and Suppression Chamber Internal Pressure	Consistent
	Drywell and Suppression Chamber Purge System	Consistent
3/4.6.2	DEPRESSURIZATION SYSTEMS	
	Suppression Chamber	Consistent
	Suppression Pool Spray	Consistent
	Suppression Pool Cooling	Consistent
3/4.6.3	CONTAINMENT ISOLATION VALVES	Consistent

4. CONCLUSION

As shown in Table I, 24 technical specification sections were compared with information in the FSAR and SER for LaSalle Unit 2. Inconsistencies were identified in two sections of the technical specifications shown in Table I. This review did not determine the significance of the inconsistency or which of the documents was in error.

5. REFERENCES

1. LaSalle County Station, Unit 2, Technical Specifications Rev. December 1983

2. LaSalle County Station, Unit 2, FSAR up to Amendment No. 63

3. LaSalle County Station, Unit 2, SER up to Supplement No. 7

NRC FORM 335 (11-81) U.S. NUCLEAR REGULATORY COMMISSION BIBLIOGRAPHIC DATA SHEET	1. REPORT NUMBER (Assigned by DDC) EGG-EA-6539
4. TITLE AND SUBTITLE Audit of Nuclear Plant Technical Specifications LaSalle County Station, Unit 2	2. (Leave blank)
	3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S)	5. DATE REPORT COMPLETED MONTH March YEAR 1984
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) EG&G Idaho, Inc. Idaho Falls, ID 83415	DATE REPORT ISSUED MONTH March YEAR 1984
	6. (Leave blank)
	8. (Leave blank)
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Division of Licensing Office of Nuclear Reactor Regulation U.S. Nuclear Regulatory Commission Washington, DC 20555	10. PROJECT/TASK/WORK UNIT NO.
	11. FIN NO. A6816
13. TYPE OF REPORT Technical Evaluation Report (TER)	PERIOD COVERED (Inclusive dates) February 13, 1984 to March 12, 1984
15. SUPPLEMENTARY NOTES	14. (Leave blank)

16. ABSTRACT (200 words or less)

This report documents the review of the LaSalle County Station, Unit 2 (LaSalle-2) Technical Specifications (T/S) to determine if selected sections of the T/S are consistent with the LaSalle-2 Final Safety Analysis Report (FSAR) as amended, and the LaSalle-2 Safety Evaluation Report (SER) as supplemented. Inconsistencies are listed in this report but no further evaluation was conducted to determine if the inconsistency was an indication of an error in any of the subject documents.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT	19. SECURITY CLASS (This report)	21. NO. OF PAGES
Unlimited	Unclassified	
	20. SECURITY CLASS (This page)	22. PRICE
	Unclassified	$

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

MAR 13 1984

Mississippi Power and Light Company
ATTN: Mr. J. B. Richard
Senior Vice President, Nuclear
P. O. Box 1640
Jackson, MS 39205

Gentlemen:

SUBJECT: REPORT NO. 50-416/84-06

On February 21-24, 1984, NRC inspected activities authorized by NRC Operating License No. NPF-13 for your Grand Gulf facility. At the conclusion of the inspection, the findings were discussed with those members of your staff identified in the enclosed inspection report.

Areas examined during the inspection are identified in the report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observation of activities in progress.

Within the scope of the inspection, no violations or deviations were identified.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosures will be placed in NRC's Public Document Room unless you notify this office by telephone within ten days of the date of this letter and submit written application to withhold information contained therein within thirty days of the date of the letter. Such application must be consistent with the requirements of 2.790(b)(1).

Should you have any questions concerning this letter, please contact us.

Sincerely,

David M. Verrelli

David M. Verrelli, Chief
Project Branch 1
Division of Project and
Resident Programs

Enclosure:
Inspection Report No. 50-416/84-06

cc w/encl:
J. E. Cross, Plant Manager
Ralph T. Lally, Manager of Quality
Middle South Services, Inc.

~~8404020091 XA~~

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

Report No.: 50-416/84-06

Licensee: Mississippi Power and Light Company
Jackson, MS 39205

Docket No.: 50-416

License No.: NPF-13

Facility Name: Grand Gulf 1

Inspection at Grand Gulf site near Port Gibson, Mississippi

Inspectors: C. Julian for	3/13/84
S. Butler	Date Signed
C. Julian for	3/13/84
J. Caldwell	Date Signed
C. Julian for	3/13/84
R. Carroll	Date Signed
C. Julian for	3/13/84
M. Hunt	Date Signed
C. Julian	3/13/84
C. Julian	Date Signed
C. Julian for	3/13/84
H. Krug	Date Signed
C. Julian for	3/13/84
N. Merriweather	Date Signed
C. Julian for	3/13/84
W. T. Orders	Date Signed
C. Julian for	3/13/84
A. Wagner	Date Signed
Approved by: [signature]	3/13/84
D. M. Verrelli, Branch Chief Division of Project and Resident Programs	Date Signed

~~8404020095 XA~~

SUMMARY

Inspection on February 21-24, 1984

Areas Inspected

This special announced inspection involved 234 inspector-hours on site in the area of verification of the accuracy of the Technical Specification.

Results

Of the areas inspected, no violations or deviations were identified.

REPORT DETAILS

1. Persons Contacted

   Licensee Employees

   *J. E. Cross, Plant Manager
   *R. F. Rogers, Assistant Plant Manager - Operations
   *C. R. Hutchinson, Assistant Plant Manager - Maintenance
   *J. W. Yelverton, Assistant Plant Manager - Support
   *J. C. Roberts, Technical Support Staff
   *F. M. Walch, Maintenance Superintendent
   *G. A. Zinke, Technical Engineering Supervisor
   *L. F. Daughtery, Compliance Superintendent
   *J. D. Bailey, Compliance Coordinator

   Other licensee employees contacted included numerous engineers, operators, mechanics, security force members, and office personnel.

   Other Organizations

   *M. G. Farschon, General Electric Site Operations Manager

   *Attended exit interview

2. Exit Interview

   The inspection scope and findings were summarized on February 24, 1984, with those persons indicated in paragraph 1 above. The Technical Specification (TS) discrepancies were described to plant management by the inspectors. NRC representatives stated that the problems found are indicative of the need for another review of Technical Specifications to find and correct any errors.

3. Licensee Action on Previous Enforcement Matters

   Not inspected.

4. Unresolved Items

   Unresolved items were not identified during this inspection.

5. Suppression Pool and Containment Spray

The inspectors compared applicable sections of the Final Safety Analysis Report (FSAR), as-built drawings, surveillance, and operating procedures and actual plant systems to Technical Specifications (TS) associated with the suppression pool and containment spray. The following are discrepancies that were identified:

a. FSAR Section 6.2.7.5 indicates that the suppression pool level indication system is made up of four level detector channels, (two detector channels per division). It also indicates that each of these channels provides a high-water-level alarm, low-water-level alarm, low-low-water-level alarm, as well as a signal to open suppression pool makeup valves.

   In actuality, there are three active level detector channels per division. Two channels are wide range and one channel is narrow range. There is also one additional channel per division which is only used for indication at the remote shutdown panel. Each wide range channel supplies input to their respective division's suppression pool makeup system in one out of two logic as well as providing a low-low-level alarm at 16'10". The narrow range channel in each division provides the divisional low-level and high-level alarms (18'5½" and 18'9", respectively). TS are written to conform to the FSAR but are not in clear agreement with the actual plant design.

b. TS 3.5.3 (ECCS), 3.6.3.1 (Depressurization Systems), 3.3.7.5 (Accident Monitoring Instrumentation), and 3.6.3.4 (Suppression Pool Makeup) all relate to required operability of suppression pool and level instrumentation. They do not recognize the difference between narrow and wide ranges and therefore do not identify what level detector channel is to be used to meet the TS operability requirement. As a result, divisional operability is left to the interpretation of the reader in the action statements as well as in the surveillance requirements.

c. In none of the above listed TS is the level instrumentation required to initiate automatic suppression pool makeup addressed as a requirement for suppression pool operability. This is an accident mitigation function and should have an associated surveillance. It would logically follow that at least TS 3.6.3.4 (Suppression Pool Makeup) should include the wide range level instrumentation as part of its operability requirement and a surveillance should be included. In TS 3.3.7.5 (Accident Monitoring Instrumentation) only two suppression pool level detectors are required, and a seven day Action Statement applies if only one is available. In reality, it appears this should read that two wide range level channels per division are required, minimum channels operable per division is one, and if only one division is operable, then the 7-day Action Statement applies.

d. By annotating what level detectors are required in the daily operating log and surveillance procedures, the license has made an effort to compensate for these unclear technical specifications. In spite of this, some problems were observed. The daily operating log indicates that for operability statement "a" of TS 3.5.3, the narrow range level detectors are to be used to verify that suppression pool level is ≥ 18'4 3/4" (Condition 1, 2, or 3); for operability statement "b", the wide range level detectors are to be used to verify that suppression pool level is ≥ 12'8" (Condition 4 or 5) since narrow range indication does not go down this far. However, wide range is calibrated for post accident temperature (170°F), thereby indicating approximately 3" higher suppression pool level than what is actually present under normal conditions. The licensee has agreed to resolve this temperature calibration issue. This will be identified as Inspector Follow-up Item (IFI) 416/84-06-01.

e. Furthermore, since the wide range indication is utilized by the licensee in conditions 4 or 5, a channel calibration per surveillance requirement 4.5.3.1.b.3 (ECCS) is required. A channel calibration is performed by surveillance procedure 06-IC-IE30-R-0001, but only for surveillance requirements 4.3.7.5 (Accident Monitoring Instrumentation) and 4.6.3.4.c (Suppression Pool Makeup). The fact that this surveillance procedure does not recognize surveillance requirement 4.5.3.1.b.3 (ECCS) further demonstrates the need for individual level instrumentation identification in these associated suppression pool TS. It is also important that all TS relating to the suppression pool cross reference each other. As it stands now, only 3.6.3.1 and 3.5.3 reference one another. The FSAR states that the level sensors are spaced 90 degrees apart around the pool. Actually, the two groups of sensors are spaced 180 degrees apart.

f. TS 4.5.3.1.a.2 contains an apparent typographical error. The licensee noted that a change request has been prepared to surveillance requirement 4.5.3.1.a.2 (ECCS) to indicate suppression pool level as 12'8", in lieu of 12'5" (IFI 416/84-06-02).

g. TS 3.6.3.1 (Depressurization Systems) and 3.3.7.5 (Accident Monitoring Instrumentation) specify suppression pool temperature requirements. There are actually installed 24 temperature detectors/alarms. (2 divisions with 6 pairs per division). The suppression pool is azimuthally divided into six sectors, with two pairs (one pair per division) in each sector. By licensee designation, 12 of these detectors are used to meet TS 3.6.3.1 and the other 12 are used to meet TS 3.3.7.5. Consequently, only 12 channels undergo the channel functional test required by surveillance requirement 4.6.3.1.c (Depressurization Systems). 4.3.7.5 (Accident Monitoring Instrumentation) does not require a functional test. Neither TS indicates

which temperature channels are to be used; therefore, leaving divisional/ sector operability to the interpretation of the reader in the Action Statements as well as in the surveillance requirements. In fact, surveillance requirement 4.6.3.1.c implies that you can use any 12 temperature channels as long as there are two channels in each sector. Since only 12 channels receive functional testing, only these 12 should be credited by TS.

TS Table 3.3.7.5-1 apparently should state as "required number of channels" 12,2/sector rather than the present 6,1/sector. Then the present statement of 6,1/sector for "minimum channels operable" would allow operation for up to 7 days in an Action Statement.

It was further observed in TS 3.6.3.1 that the combining of action statements on suppression pool level and temperature instrumentation with the use of "and/or" was very ambiguous.

h. TS 3.6.3.2 (Containment Spray) contains an error and the licensee stated that a change has been prepared to operability statement 3.6.3.2.b to indicate the use of a "RHR" heat exchanger, in lieu of a "SSW" heat exchanger (IFI 416/84-06-03). Another inconsistency in surveillance requirement 4.6.3.2.b was pointed out by the inspectors.

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In order to demonstrate operability of containment spray, this surveillance requires verification that each RHR pump develops a flow of at least 5650 GPM while recirculating water through the RHR heat exchanger to the suppression pool. This is accomplished by surveillance procedure 06-OP-1E12-Q-0023, where the same recirculation flow verification is used to determine LPCI and suppression pool cooling operability. However, surveillance requirements 4.5.1.b.2 (LPCI) and 4.6.3.3.b (Suppression Pool Cooling) specify a recirculation flow of at least 7450 GPM. The FSAR indicates that containment spray flow emitting from the spray nozzles into the containment is 5650 GPM. This would imply that a RHR pump flow capability of 7450 GPM is reduced to 5650 GPM after passing through the piping and containment spray nozzles. Therefore, one would suspect that surveillance requirement 4.6.3.2.b (Containment Spray) should also require a RHR recirculation flow acceptance criteria of at least 7450 GPM. At the time of the inspection, the licensee was unable to provide their spray flow analysis to justify the lower RHR recirculation flow of 5650 GPM.

In regards to an inoperable train, Action statements of TS 3.5.1 (LPCI) and 3.6.3.3 (Suppression Pool Cooling) allow for seven day continued operation when only one train is available. Since containment spray is more important, i.e., has less redundancy, action statement 3.6.3.2.a (Containment Spray) only allows a 72 hour Action period when one train is inoperable. A review of the RHR Pump operability data sheets in surveillance procedure 06-OP-1E12-Q-0023 revealed an allowance of 96 hours to analyze test results. In essence, this allows an additional

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96 hours possible delay to the Action periods discussed above. Licensee representatives agreed to review this matter and make appropriate changes to the surveillance procedure (IFI 416/84-06-04).

6\. a. TS 3/4.6.6.2 - Secondary Containment Automatic Isolation Dampers/Valves

The inspector compared the TS list of automatic valves and dampers to the licensee's surveillance procedures. The completed results of the most recent surveillance on secondary containment isolation were reviewed to see that the valve lists and identification are compatible. Approximately 5% of the valves and dampers were examined in the plant by the inspector. No discrepancies were identified.

The inspector asked licensee representatives if an actual plant walk-down had been conducted by the licensee to verify the accuracy of the TS lists of primary, drywell, and secondary valves. Licensee representatives stated that walkdowns were done at various times to resolve specific questions, but no comprehensive effort could be identified which had as its objective the verification of the TS tables. The inspector stated that, although this is not a regulatory requirement, it would seem to be a prudent action to confirm TS accuracy. Licensee representatives agreed to consider further action (IFI 416/84-06-05).

b. TS 3/4.6.6.3 - Standby Gas Treatment

The inspector reviewed the surveillance procedure for SGTS. The TS surveillance requirements and the implementing procedures appear adequate to ensure SGTS reliability. The inspector walked down the majority of the SGTS hardware in the plant to ensure that the hardware is compatible with the TS. No discrepancies were observed.

c. TS 3/4.6.7.1 - Hydrogen Recombiner

The inspector examined the two hydrogen recombiner systems installed in the plant to ensure compatibility with the TS. The completed results of the preoperational tests of this equipment were reviewed to ensure that the recombiners are capable of performance described in the surveillance section of the TS. No discrepancies were observed.

7\. TS 3/4.8 - Emergency Power Supplies

The inspectors selected several sections of the TS and the corresponding surveillance procedures for examination to verify the adequacy of the procedures and the TS as they relate to the existing equipment. The following TS and surveillance procedures were examined and evaluated.

TS Sections

3/4.8.1	AC Sources - Operating
3/4.8.2	DC Sources - Operating
3/4.8.3	Onsite Power Distribution Systems (Operating)
3/4.8.4	Electrical Equipment Protective Devices Primary Containment Penetration Conductor Protective Devices - Motor Operated Valve Thermal Overload Protection Reactor Protection System Electric Power Monitoring

Surveillance Procedures

06-OP-1R20-W-0001	Plant AC and DC Electrical Power Distribution Weekly Lineup
06-EL-1L51-R-0001	125V Battery Charger Capability Test
06-IC-1C71-SA-1001	RPS Electrical Protection Assembly Channel Functional Test
06-EL-1C71-R-0012	RPS Electrical Protection Assembly Calibration
06-EL-1L11-O-0001	125V Battery Capacity Discharge Test
06-EL-1R65-Q-1001	MOV Thermal Overload Protection Device
06-EL-1R65-R-0001	MOV Thermal Overload Protection Device

As a result of this review, the following discrepancies were identified:

a. Surveillance procedure 06-EL-1L51-R-0001 appears to be inadequate in that the battery chargers are never tested at the equalizing voltage (140 VDC ± 1 volt). The chargers are only tested at 105 volts at 400 amperes for two (2) hours. In addition, in the battery discharge test, there is no time limitation specified for when the batteries must be recharged to full capacity (IFI 416/84-06-06).

b. An apparent typographical error was found in TS Table 3.8.4.2-1. The B designation was omitted from valve number QSP415189B. Licensee representatives have since stated informally that the TS is correct. This will be confirmed during a future inspection.

c. TS requirement 3.8.4.3 appears to be inappropriate for the way the RPS electrical power monitoring assemblies (EPAs) are designed. Two EPAs are in series which means both units must be operable to supply power to the RPS bus. The Action statement in the TS requiring only one (1) EPA unit to be restored to service when two are inoperative does not seem appropriate for the circumstance. There is no provision for manual bypass of the individual EPA units.

   Surveillance procedure 06-IC-1C71-SA-1001 appears inadequate in that it only requires testing of the EPAs that are not providing power to the Reactor Protection System (RPS) bus. The procedure does not assure that the EPAs associated with the normal power supply (MG sets) will be tested during the six month surveillance test as required by TS Section 4.8.4.3.a (IFI 416/84-06-07).

The NRC inspectors also performed walkdowns of the systems identified above to randomly verify that equipment described in the TS was actually installed in the plant. All equipment examined in the plant was found to be properly identified in the TS with the exception of the items discussed above.

8. ECCS Systems and Actuation Instrumentation

a. TS 3/4.5.1 ECCS - Operating

The requirements for ADS operability contained in paragraphs 3.5.1.a.3 and 3.5.1.b.2 were reviewed. The TS paragraphs require "at least 7 operable ADS valves". This number appears to be incorrect. The Safety Evaluation Report page 6-22 states that the ADS employs eight of 20 SRVs. The action statement paragraph e.1 allows the operation up to 14 days with only six ADS valves operable, and up to 12 hours with five or less ADS valves. This appears to be an unacceptable TS (IFI 416/84-06-08).

A review of the paragraph 4.5.1.b pump testing criteria was conducted. Significant inconsistencies were noted in the pump flow characteristics for the following pumps. The TS for High Pressure Core Spary requires at least 7115 gpm with 182 psid, while the SER page 6-21 states 7115 gpm with 540 psid, and the FSAR Figure 6.3-2 lists 7115 gpm with approximately 387 psid. The TS for Low Pressure Core Spray requires at least 7115 gpm with 261 psid, while the SER page 6-22 states 7115 gpm with 340 psid and the FSAR lists 7115 gpm with approximately 311.6 psid. The TS required flows appear considerably less conservative than either the SER or FSAR (IFI 416/84-06-09).

b. TS 3/4.3.3 Emergency Core Cooling System Actuation Isolation

The inspector verified the incorporation of the following instrument surveillances of TS Tables 3.3.3-1, 3.3.3-2, and 4.3.3.1-1 into the plant's surveillance program.

LPCI Pump A Start Time Delay Relay
ADS Times
Drywell Pressure High
Reactor Vessel Water Level - Low, Low, Level 2

The following surveillance procedures were reviewed to ensure the required TS frequencies, trip setpoints, and allowable values were correctly incorporated.

06-IC-1B21-R-0012, Rev. 22, Reactor Vessel Water Level Calibration
06-IC-1B21-M-1010, Rev. 21, TCN9 Reactor Vessel Water Level (HPCS)
06-EL-1B21-M-0001, Rev. 21, TCN3 ADS Times Functional Test and Calibration

06-OP-1000-D-0001, Rev. 20, TCN27 Daily Operating Log (Items 64 & 16)
06-IC-1B21-R-0009, Rev. 21, TCN3 Drywell High Pressure Calibration (ECCS)
06-IC-1B21-M-1011, Rev. 20, TCN4 Drywell High Pressure (HPCS) Functional Test
06-EL-1E12-M-0001, Rev. 22, RHR Pump Start Time Delay Relay Functional
06-EL-1E12-M-0001, Rev. 21, RHR Pump Start Time Delay Relay Calibration

The following Drywell Pressure Transmitters and Reactor Vessel Level Transmitters were reviewed for proper field installation in accordance with the as-built drawing and the piping and instrumentation diagrams. Logic diagrams were reviewed for actuation of the appropriate equipment. No deficiencies were noted.

Division I	PT	N094A
Division I	PT	N094E
Division II	PT	N094B
Division II	PT	N094F
Division III	PT	N067C
Division III	PT	N067G
Division III	PT	N067L
Division III	PT	N067R
Division I	LT	N091A
Division I	LT	N091E
Division II	LT	N091B
Division II	LT	N091F
Division III	LT	N073C
Division III	LT	N073G
Division III	LT	N073L
Division III	LT	N073R

The licensee has previously identified a problem on instruments which utilize atmospheric pressure as one side of a differential pressure detector instrument. Due to a possible low atmospheric pressure condition around the plant, certain detectors may be as much as .5 psig nonconservative. This includes drywell pressure and containment spray. The licensee has not yet submitted all the appropriate changes at this time (IFI 416/84-06-10).

9. Drywell and Primary Containment Integrity

An inspection was performed of the following sections of the Grand Gulf TS:

SECTION	SUBJECT	PAGES
3/4 6.1.1	Primary Containment Integrity	3/4 6-1
3/4 6.1.2	Containment Leakage Rates	3/4 6-2, 3, 4

3/4 6.1.3	Containment Air Locks	3/4 6-5, 6
3/4 6.1.4	MSIV Leakage Control System	3/4 6-7
3/4 6.1.6	Containment Structural Integrity	3/4 6-9
3/4 6.1.7	Containment Internal Pressure	3/4 6-10
3/4 6.1.9	Containment Purge System	3/4 6-12

Emphasis was placed on the following specifics:

(1) Literal correspondence between the TS, and the installed hardware configuration.

(2) Adequacy and completeness of the surveillance requirements.

(3) Review of associated surveillance procedures and results generated by their execution.

(4) Adequacy and completeness of the Action Statements.

(5) Familiarity of licensee personnel with the TS and the associated hardware systems and testing requirements.

The following discrepancies were identified:

a. Surveillance Requirement 4.6.1.4 concerns the operability of each MSIV leakage control subsystem. Item C address the functional testing of the subsystem heaters but does not acknowledge that there are no heaters on the outboard subsystem; whereas there are four on the inboard subsystem. Section 6.7 of the FSAR reveals that no heaters are required in the outboard system, which is not identical to the inboard system in a number of aspects. Additionally, licensee surveillance procedures accurately reflect the existing hardware configuration. Clarification of the wording of the TS will resolve the ambiguity (IFI 416/84-06-11).

b. Surveillance Requirements 4.6.1.1.a requires a leak rate retest of the equipment hatch seals every time each penetration subject to a Type B test, except the containment air locks, is reclosed.

   This is not what was intended as it would require a retest of the equipment hatch seals following the opening of Type B pentration areas such as:

   (1) electrical penetrations

   (2) ECCS test return line orifice plate

   (3) fuel transfer tube

Surveillance Requirement 4.6.1.1.b is vague as to what must be secured in position, and how. Correction of the wording in the TS will resolve these ambiguities (IFI 416/84-06-12).

10. TS 3/4.3.2 Isolation Actuation Instrumentation

TS 3/4.3.2 Isolation Actuation Instrumentation was reviewed to determine if the requirements entailed therein are clear, if the LCOs are realistic, if the channels and trip systems appear technically adequate, if there are procedures for performing the surveillances, and if those requirements can be performed.

Seventeen applicable surveillance procedures were analyzed for technical adequacy and incorporation of TS acceptance criteria. Ten channels and associated trip systems were analyzed through examination of electrical prints, logic diagrams, and system descriptions for technical adequacy. The review revealed that TS 3/4.3.2 appeared technically adequate, the requirements realistic, and there were procedures for performing those selected requirements reviewed.

The procedures reviewed, with those exceptions to be detailed, appeared adequate. The procedures reviewed included but were not limited to:

06-IC-1B21-M-1004

06-IC-1B21-M-1010

06-IC-1C71-M-0001

06-IC-1B21-M-1004

06-IC-1E31-M-0003

06-OP-1000-D-0001

06-IC-1E31-M-1001

06-OP-1G33-M-0002

06-IC-1E31-M-0023

06-IC-1321-M-1003

Of those procedures, some discrepancies were identified in operation surveillance procedure 06-OP-1000-D-0001, which entails the operations' semi daily surveillance (channel checks) as required by TS Table 4.3.2.1-1. The item numbers, detailed below refer to the TS table specific requirement, by line number. The discrepancies are as follows:

a. With regard to items 1.b through 1.g and 5.m it was observed that the procedure line item (as referred to by the TS cross reference index) does not conform to the TS requirement. This appears to be an error in the TS cross reference document.

b. Line item 63 of procedure 06-OP-1000-D-0001 stipulates a channel check of slave trip unit B21-LS-693 to level instrument B21-LIS-N691 at $\leq$ 53.5". The slave trip unit referenced on the semi-daily surveillance sheet has undergone a station modification such that it no longer is slave unit to instrument B21-LIS-N691, as the procedure indicates, but now is slave to instrument B21-LIS-N695. There are three problems associated with this issue, of serious concern.

   (1) The station modification was completed, according to the licensee, in December 1983; however, the procedure has yet to be changed to reflect the modification.

   (2) The operations staff, if they were to be following the procedure, would be noting in mode 4 that slave unit B21-LS-693 is not in alarm as it should be since the master trip unit B21-LIS-N691 is tripped and is the procedure referenced master trip for that slave. They are not noting this fact.

   (3) The operations staff has been ignoring the procedural requirements simply because they know the modification, discussed above, is installed and as such that slave trip unit should not be in alarm. The operations staff however, has not in the period since December initiated a procedure change to reflect the station modification.

c. Further, TS item 4.3.2.1-1.f requires that a semi-daily channel check be performed on high drywell pressure, ECCS, Division 3. The TS cross reference index refers to line item 15 of procedure 06-OP-1000-D-0001 for the performance of that required surveillance. That line item number appears to test reactor vessel level 2 and 8.

d. Line items 15 and 63 (reactor vessel level 8) require a semi-daily channel check of those applicable instruments at $\leq$ 53.5". However, the instruments referred to in the procedure for those channel checks are calibrated for elevated temperatures, and in modes 4 and 5, are pegged high such that the required channel check can never be satisfactorily performed. Here again, although operations staff has known of this inadequacy, no procedure change has been implemented to resolve the inadequacy, i.e., refer to an instrument which reads correctly in modes 4 and 5.

In summary, the TS (3/4.3.2) appears to be adequate although there are some inadequacies in the implementation of the requirements. Licensee representatives agreed to review and correct as necessary the semi-daily log sheet and the TS cross reference document (IFI 416/83-06-13).

11. TS 3.6.4 - Containment and Drywell Isolation Valves

The inspector reviewed TS 3.6.4 "Containment and Drywell Isolation valves" and Tables 3.6.4-1 to verify that the licensee has adequately identified all primary containment and drywell penetrations and associated isolation valves and included them in the TS. In addition, the inspector reviewed the limiting condition for operation and surveillance requirement for TS 3.6.4 to ensure that they were appropriate and being properly implemented by the licensee.

The inspector reviewed Final Safety Analysis Report (FSAR) Table 6.2-44 "Containment Isolation Valve Information" and Figures 6.2-76 thru 80 "Containment Leak Rate Test System" and compared them to TS Table 3.6.4-1, plant surveillance procedures and selected "as built" penetrations observed in the plant. No discrepancies were identified. The inspector reviewed the following plant surveillance procedures:

06-OP-1M10-M-0001	"Containment and Drywell Penetration Isolation Monthly Check"
06-OP-1M10-C-10C1	"Containment and Drywell Penetration Isolation Cold Shutdown Check"
06-OP-1B21-R-0006	"Containment, Drywell and Auxiliary Building Isolation Valves Functional Test"
06-OP-1B21-C-0003	"Nuclear Boiler Valve Operability"

The procedures were reviewed to determine if they adequately fulfilled the associated surveillance requirements and corresponded to the TS Table 3.6.4-1 of isolation valves. Some minor discrepancies were identified and resolved. No valves were identified that were not covered by procedure or TS.

The inspector toured the Unit 1 Containment and Drywell and randomly selected approximately 45 isolation valves and verified that they were included in TS Table 3.6.4-1 and plant surveillance procedures. No discrepancies were identified. The comments of paragraph 6.a and the IFI identified there also include this large group of valves.