APPENDIX

U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-458/85-52 Construction Permit: CPPR-145

Docket: 50-458

Licensee: Gulf State Utilities
P.O. Box 2951
Beaumont, Texas 77704

Facility Name: River Bend Station

Inspection At: St. Francisville, Louisiana

Inspection Conducted: July 8-12, 1985

Inspectors: ____________________ 8/23/85
C. C. Harbuck, Reactor Inspector Date
Project Section A, Reactor Projects Branch

____________________ 8/23/85
T. A. Flippo, Resident Inspector, Waterford-3 Date
Project Section C, Reactor Projects Branch

Approved: ____________________ 8/23/85
J. P. Jaudon, Chief, Project Section A Date
Reactor Projects Branch

____________________ 8/23/85
G. L. Constable, Chief, Project Section C Date
Reactor Projects Branch

Inspection Summary

Inspection Conducted July 8-12, 1985 (Report 50-458/85-52)

Areas Inspected: Routine, unannounced inspection of Emergency and Abnormal Operating Procedures, IE Bulletin followup, Post TMI-Action Plan Requirements followup, and the corrective actions taken for deficiencies previously reported pursuant to 10 CFR Part 50.55(e). The inspection involved 168 inspector-hours onsite by two NRC inspectors and three NRC consultants.

Results: Within the four areas inspected, no violations or deviations were identified.

DETAILS

1. Persons Contacted

Gulf State Utilities

*W. J. Cahill, Senior Vice President
*T. C. Crouse, Manager Quality Assurance
*M. W. Henkel, Engineer-Nuclear Licensing
*K. E. Suhrke, Manager Project Planning & Coordination
*S. R. Radebaugh, Assistant Superintendent, Startup and Test
*C. L. Ballard, Project Supervisor
*R. E. Turner, Quality Assurance Engineer
*J. E. Spivey, Operations Quality Assurance Engineer
*D. R. Gipson, Assistant Plant Manager, Operations, Chemistry, and Radioactive Waste
*G. V. King, Plant Services Supervisor
*T. F. Plunkett, Plant Manager
*P. F. Tomlinson, Director, Operations Quality Assurance
*G. R. Kimmell, Supervisor Operations Quality Assurance
*L. A. England, Supervisor-Licensing
*R. D. Ruby, Fire Protection Engineer
*C. D. Redding, Quality Assurance Engineer-Operations
*R. W. Helmick, Director-Projects
*T. W. Overlid, Process Systems Supervisor
*P. E. Freehill, Superintendent Startup and Test
*A. D. Kowalczuk, Assistant Plant Manager
*P. J. Dautel, Licensing Staff Assistant
*B. E. Hey, Engineer
T. Bradish, Licensed Operator

Stone and Webster

*W. T. Tucker, Assistant to Superintendent of Engineering
*W. I. Clifford, Resident Project Manager
*R. L. Spence, Resident Quality Control Manager
*H. E. Stubbs, Supervisor Field Quality Control

*Denotes attendance at exit meeting held July 12, 1985.

2. Licensee Actions Taken in Regards to Previous Inspection Findings

   a. (Closed) Violation (8520-04) "Z-series Engineering Design and Coordination Reports - Implementation." The NRC inspector reviewed the licensee's response letter to the NRC, dated June 7, 1985, concerning this violation, and verified that the corrective actions

stated therin were as stated. The NRC inspector also found that the use of this type of E&DCR had been discontinued and concluded no further implementation problems were likely to have occurred. This item is closed.

b. (Closed) Open Item (8524-01) "Independent Safety Engineering Group." Based on the findings discussed in paragraph 5.k of this report, this item is closed.

c. (Closed) Open Item (8516-01) "Emergency and Abnormal Operating Procedures." Based on the findings discussed in paragraph 6 of this report, this item is closed.

3. Construction Deficiency Correction Followup Inspection

The purpose of this inspection was to determine the adequacy of corrective actions taken by the applicant on construction deficiencies identified to the NRC pursuant to the requirements of 10 CFR 50.55(e). The following attributes were checked:

- That the deficiency was properly determined to be "Reportable" or "Non-Reportable" as defined in 10 CFR 50.55(e)(1);
- If deemed "Reportable," that the applicant submitted a written report containing the information required by 10 CFR 50.55(e)(3) to the NRC within 30 days (or an interim report containing all available information, together with a statement as to when a complete report will be filed);
- That the specific hardware (or administrative) deficiency had been corrected such that it no longer exists;
- That adequate permanent corrective action was taken to preclude a repeat of the deficiency in the future;
- That sufficient documentation existed containing a clear description of the deficiency, the cause(s), an analysis of the safety implications, and what remedial actions were taken; and
- That the final report submitted to the NRC contained sufficient information to permit analysis and evaluation of the deficiency and the corrective action.

The NRC inspectors verified that the corrective actions had been completed on "Reportable" deficiencies. Most "Non-Reportable" deficiencies were corrected as of the period of this inspection; however, work completion was not considered a prerequisite for NRC closure since the deficiencies were being tracked by the applicant's internal system. The closure of a "Non-Reportable" deficiency report (DR) constitutes NRC concurrence that the item was not reportable as defined in 10 CFR 50.55(e)(1), and that adequate corrective action appeared to have been implemented.

a. The following "Reportable" DRs were reviewed by the NRC inspectors and verified satisfactorily completed in terms of the attributes listed above. In addition, the corrective actions were satisfactorily completed and there are no further questions or comments. Therefore, these reported deficiencies are considered closed.

DR NUMBER	SHORT TITLE OR SUBJECT	INITIAL NOTIFICATION	GSU FINAL REPORT LETTER
312	Abnormal wear on turbine shaft on Diesel Generator B	06-07-85	RBG-21384/06-24-85
194	Jacket water pump for standby diesel generator change in impeller material and shaft modification	05-04-84	RBG-19015/09-24-84
258	Excessive leakage of containment isolation check valves. (Note: As of the date of the final report all valves had been repaired except 1B21AOVF032A. This valve has since been reworked and tested acceptable on 5/23/85. Test report added to DR 258 to complete.)	10-19-84	RBG-21147/05-29-85
160	Transformer cable lead lugs	01-12-84	RBG-19654/12-07-84
259	Split terminal blocks manufactured by Underwriters Safety Device Co.	10-19-84	RBG-19831/01-03-85
273	Pressure transmitter environmental zone relocation	12-11-84	RBG-21385/06-13-85
289	HPCS diesel generator voltage regulator	03-06-85	RBG-20695/04-08-85

b. The following "Non-Reportable" DRs were reviewed and are considered closed.

DR NUMBER	SHORT TITLE OR SUBJECT	INITIAL NOTIFICATION	GSU FINAL REPORT LETTER
296	Linear indications in the areas of longitudinal weld seams of structural tubing.	03-28-85	RBG-21421/06-27-85

307	Pipe fittings manufactured by Bonney Forge that lacked the chemical overcheck requirement of paragraph NCA-3867.4(e)(2)	05-17-85	RBG-21452/07-05-85

4. IE Bulletin Followup Inspection

The NRC inspectors conducted a followup inspection of applicant actions taken on selected IE Bulletins. The objectives of this inspection were to ascertain whether the information submitted by the applicant in response to the IE Bulletin sent for action was technically adequate, represented the action taken by the applicant.

The attributes inspected included verification of the following:

- The written response was within time period stated in the bulletin;
- The written response included the information required to be reported;
- The written response included adequate corrective action commitments based on information presented in the bulletin and the applicant's response;
- Corporate management forwarded copies of the written response to appropriate onsite management representatives;
- Information discussed in the applicant's written response was accurate; and
- Corrective action taken by the applicant was as described in the written response.

a. The following IE Bulletins requiring an applicant response were reviewed by the NRC inspectors and have met the required objectives and the inspection attributes listed above. Therefore, the following IE Bulletins are considered closed.

(1) IE Bulletin 77-05, 77-05A, "Electrical Connector Assemblies."

ISSUE: These bulletins identify certain pin and socket connectors which were found to fail under the anticipated environmental conditions following an accident.

FINDINGS: This item was kept open on NRC Inspection Report 85-30. GSU has provided the documentation showing adequate qualification testing necessary to close this bulletin.

(2) IE Bulletin 77-08 requested the licensee to survey their facility plans to ensure prompt emergency ingress into electrically locked safety-related areas by essential personnel and unimpeded emergency egress from all areas so as not to degrade personnel safety.

ISSUE: At one nuclear plant a loss of offsite power resulted in the scram of the reactor. Electrically locked doors to vital areas failed due to lack of auxiliary power. This caused the delay of operations personnel into several safety-related areas. Concurrent with the above situation, other employees were isolated without an adequate emergency escape route available to them due to the failed electrical locking devices.

FINDINGS: GSU's letter dated February 28, 1978 (RBG-4990), was in response to the bulletin. The River Bend Station Physical Security Plan has been written and was approved by the NRC. Section 6.6.2.2. and 7.1.2 (safeguards information) fully addresses the concerns of Bulletin 77-08. Furthermore, Procedure PSP-4-071, "Security Barrier Assessment and Intrusion Detection System," ensures the operability of the security doors at RBS.

(3) IE Bulletin 79-02, "Pipe Supports Base Plate Design Using Concrete Expansion Anchor Bolts."

ISSUE: The bulletin identified a problem with failures of pipe supports utilizing base plates secured by use of concrete expansion anchor bolts. Failures were attributed to inadequate design analysis of base plates and improper installation of anchor bolts.

FINDINGS: A review of GSU's actions as stated in their response letters RBG-20872, RBG-32806, and RBG-32807 was performed. RBS original design utilized embedded plates for pipe support attachment. The use of supports incorporating base plates was to be optional at later stages of construction. At the time this bulletin was issued, GSU had not installed any supports utilizing base plates. The necessary controls to assure compliance with the intended requirements of this bulletin were inplace prior to installation of base plate type pipe supports. These controls included computer code design analysis for base plates, installation procedures for concrete anchor bolts, and commercial test data for anchor bolt actions and responses. Therefore, the NRC inspector concluded that GSU has adequately addressed the concerns of this bulletin.

(4) IE Bulletin 83-08, "Electrical Circuit Breakers With an Undervoltage Trip Feature in Use in Safety-Related Applications Other Than the Reactor Trip System."

ISSUE: To assure proper operation of circuit breakers with undervoltage trip attachments being used in safety-related applications other than reactor trip breakers.

FINDINGS: RBS has no General Electric AK-2, Westinghouse Type DB, or Westinghouse Type DS circuit breakers in an undervoltage trip feature in use in any safety-related applications. Other similar circuit breakers have been performed and surveillance programs, preventive and corrective action measures identified in STPs 508-1600 and 508-1700, and Technical Specifications 4.8.4.2 and 4.8.4.3b. Applicable GSU response letters are RBG-17,445, RBG-20,634, and RBG-21,438.

(5) IE Bulletin 84-02, "Failure of General Electric Type HFA Relays in Use in Class IE Safety Systems."

ISSUE: Failure of General Electric Type HFA relays have been reported. The latest failures indicate that this model relay is being used in some safety-related systems. Facilities are to identify these relays in their safety systems and plan upgrades where applicable.

FINDINGS: On July 3, 1985, GSU made a final response (RBG-21446) to Bulletin 84-02. All safety-related HFA relays utilized at RBS have been identified and upgraded. Administrative controls, Quality Assurance Inspection Plan QAI 2.18, and the excluded equipment list, have been established to ensure that nonsafety-related HFA relays are not used in safety-related applications, and that spare parts for these relays will only be of the type applicable for safety-related applications.

b. The following IE Bulletins were determined not to have required a response from the applicant but there was documented evidence that each IE Bulletin was reviewed and utilized by GSU for useful information that might be applicable in the future.

(1) IE Bulletin 78-08, "Radiation Levels From Fuel Element Transfer Tubes."

ISSUE: Inadequate shielding design and lack of proper administrative controls during usage of spent fuel transfer systems at nuclear facilities.

FINDINGS: A review of GSU's actions and their response letter, RBG-19453, including referenced documentation was performed. GSU has adequately addressed the concerns of this bulletin concerning design of shielding and administrative controls for usage of their "IFTS" system.

(2) IE Bulletin 79-08, "Events Relevant to Boiling Water Power Reactors Identified During Three Mile Island Incident."

ISSUE: This bulletin identified improvements which could be made by the licensee as a result of lessons learned from the Three Mile Island incident.

FINDINGS: RBS has addressed each of these areas in the specific responses to NUREG-0737 items which are found in the FSAR.

(3) IE Bulletin 79-13, "Cracking in Feedwater System Piping."

ISSUE: The bulletin identified a problem with cracks developing in feedwater piping systems at PWR plants.

FINDINGS: A review of GSU's actions and their response letter RBG-21335, including referenced documents, was performed. The NRC inspector concluded that GSU has adequately addressed the concerns of this bulletin. The bulletin addressed a generic problem with PWR design configuration that is not present in the RBS plant. GSU has researched the potential occurrence of this problem by verifying that they are in compliance with NUREG-0619 which addresses BWR plants.

(4) IE Bulletin 79-18, "Audibility Problems Encountered on Evacuation of Personnel from High Noise Areas."

ISSUE: Not all personnel were able to hear the evacuation announcement made over the public address system while working in a high noise area.

FINDINGS: E&DCR C-27,870 was added to Stone & Webster Specification 248.000 to provide criteria for audibility of alert signals and the subsequent testing of high noise levels to insure evacuation signals can be heard.

(5) IE Bulletin 79-19, "Packaging Low Level Radioactive Waste for Transportation and Burial."

ISSUE: This bulletin identifies the concerns over the serious and repeated disregard for rules governing the shipment of low level radioactive waste to burial facilities and noncompliance to DOT and NRC requirements. Action to be taken by the licensee was in nine parts to correct the problem.

FINDINGS: On April 3, 1985, GSU responded to the bulletin (S-CRB 6489) informing the NRC that action was competed on Parts 1 through 7 and that Items 8 and 9 did not apply at this time.

(6) IE Bulletin 79-20, "Packaging Low Level Waste for Transportation and Burial."

ISSUE: This bulletin identifies the concerns over the serious and repeated disregard for rules governing the shipment of low level radioactive waste to burial facilities and noncompliance to DOT and NRC requirements. Action to be taken by the licensee was in eight parts to correct this problem.

FINDINGS: On April 3, 1985, GSU responded to the bulletin (S-CRB 6490) informing the NRC that action was completed on Parts 1 through 7. Part 8 did not apply at this time.

(7) IE Bulletin 79-24, "Frozen Lines."

ISSUE: The bulletin identified a generic problem with exposed process, instrument, and sample lines freezing during extremely cold weather. Problem areas included improper heat trace design and inadequate control systems.

FINDINGS: A review of GSU's actions and their response letters S-CRB-6902 and RBG-32799 was performed. GSU has established a system for identification of potential problem lines and has established a control system within the operating procedures to assure that heat traced lines are actuated and verified operable during cold weather.

(8) IE Bulliten 80-15, "Possible Loss of Emergency Notification System (ENS) With Loss of Offsite Power."

ISSUE: Bulletin 80-15 addresses the concern of loss of offsite power causing loss of communications between a facility and the NRC Operations Center.

FINDINGS: There is documented evidence that GSU reviewed and used the information in this bulletin in the design of their ENS system.

(9) IE Bulletin 83-04, "Failure of the Undervoltage Trip Function of Reactor Trip Breakers."

ISSUE: This bulletin identified a potential problem with failure of General Electric AK-2 type circuit breakers equipped with undervoltage (UV) trip attachments. The major concern is the utilization of this type of breaker in the reactor trip functions.

FINDINGS: GSU has done an extensive review of the applications for this type of circuit breaker at RBS for both safety-related and reactor trip circuitry. There are no GE Type AK-2 circuit breakers with IV trip features utilized in any safety-related applications. (Ref. closed Bulletins 79-09 and 83-08.)

5. Post TMI-Action Plan Requirements

The objective of this portion of the inspection was to determine the status of Post TMI Action Plan requirements.

a. Item II.D.1, Over Pressure Protection - This item is to ensure that the reactor coolant pressure boundary (RCPB) is provided with a pressure relief system to prevent pressure within the RCPB from rising beyond 110% of the design value.

Findings - Review of FSAR Section 5.2.2 and plant inspection determined that the relief and safety valve system, committed to, exists in the plant. Sixteen valves of the balanced, spring loaded type have an auxiliary power-actuated device that allows opening of the valves even when the pressure is less than the safety-set pressure of the valves. The valves are the CROSBY type. Qualifying tests were made by the Wylie Labs and reported in NEDE-49882-P (analysis of generic BWR safety/relief valve operability test results). This item is considered closed.

b. Item II.D.3, Valve Position Indication - This item is to ensure that the RBS design includes a reliable indication of flow in each safety/relief valve (SRV) and a common annunciator located in the control room.

Findings - Review of applicable FSAR sections and plant inspection determined that two types of SRV flow indicators are installed in the control room.

(1) There is a red light on the console for each of the 16 SRVs indicating that the valve is not closed when lighted. In addition, located near the console is a panel of 10 LEDs for each SRV indicating the percentage open from 0-100%. The above indicators are operated by acoustic sensors located downstream of the SRVs.

(2) A multi point recorder records the temperature of a thermocouple located just downstream of each SRV. The common annuciator for SRV opening is connected to the set point of the recorder.

This item is considered closed.

c. Item II.F.1, Attachments 4, 5, and 6 - Containment Pressure, Water Level, and Hydrogen Monitors - This item requires that continuous indication be provided in the control room of containment pressure, containment water level, and containment hydrogen concentration.

Findings - Instrumentation required by this item is installed. Preoperational testing has been performed and test deficiencies are resolved. The applicant has satisfactorily complied with the requirements of this item. This item is considered closed.

d. Item II.F.2, Instrumentation for Detection of Inadequate Core Cooling - Item II.F.2 requires the applicant to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC).

Findings - Instrumentation at the RBS provides the operators with redundant full range, easy-to-interpret indications of trend toward, and having reached ICC. The fuel zone water level indication has been incorporated into training. This item is considered closed.

e. Item II.K.3.27, Provide Common Reference Level for Vessel Level Instrumentation - This item requires all reactor water level instrumentation to be referenced to the same point to prevent operator confusion.

Findings - The applicant has made and documented the changes necessary to comply with this item, and has also incorporated the necessary changes in training. This item is considered closed.

f. Item II.K.3.15, Modify Break Detection Logic to Prevent Spurious Isolation of High Pressure Coolant Injection and Reactor Core Isolation Cooling - This item identified a need for modification of the pipe break detection circuitry of the HPCI/RCIC systems. The intended modification was needed to limit inadvertent system isolation due to the pressure spike which accompanies system startup. To resolve the concerns of this action item, GSU committed to modify the existing circuitry by installing a time delay relay configuration.

Findings - The system modification was verified complete by review of design documentation, acceptable installation, system testing, and physical inspection in the field. Design documents reviewed

included the RCIC design specification, GE 22A3124BC, "Elementary Logic Diagram 828E539AA," and the RCIC Flow Diagram FSK 27-6C. Test data reviewed included Loop Calibration Reports 1.IL-ICS.D2D and 1.IL-ICS.021 and preoperational Test Procedure 1-PT-209. This item is considered closed.

g. Item II.K.3.13, Separation of High Pressure Coolant Injection and Reactor Core Isolation Cooling System Initation Levels - This item required analysis and modification of the control logic for operation of the reactor core isolation cooling (RCIC) and the high pressure coolant injection (HPCI) systems. Due to the different flow characteristics and functional operation, including concerns for vessel damage because of thermal shock, several modifications were suggested. GSU review of the action item concerns and subsequent analysis determined that two modifications were required. The RCIC system would be modified to automatically restart following a trip of the system at high reactor vessel water level and that the system would automatically restart on low water level.

Findings - To assure completion of the intended modifications, a review of the RBS design disclosure documentation and inspection of the installation was conducted. The proposed modifications have been incorporated into the necessary design documents, including the following: Design Specification GE-22A3124 BC, Elementary Logic Diagram 828E539AA, P&ID 796E726, and Functional Control Diagram 762E2977AA. Field modification was performed per Engineering and Design Report C-52,121. Both modifications were functionally tested and accepted per Preoperational Test Procedure 1-PT-209. This item is considered closed.

h. Item II.K.1.22, Auxiliary Heat Removal System Procedures - This item addresses the need to ensure that procedures exist for proper operation of auxiliary heat removal systems and to ensure implementation of licensee commitments for auxiliary heat removal.

Findings - Prior inspection activities (NRC Inspection Report 50-458/85-12) verified that RBS residual heat removal system and procedures were adequate to fulfill the commitments of FSAR Appendix IA. At that time the item was held open pending completion of modification and testing of the reactor core isolation cooling system (Ref. II.K.3.13). The modifications to the RCIC system have been completed and tested and TMI Action Item II.K.3.13 has been closed. This item is considered closed.

i. Item I.A.1.1, Shift Technical Advisor - This item requires implementation of the shift technical advisor (STA) position and provides guidelines for the education, training, and experience needed by personnel filling this position.

Findings - This item was reviewed in NRC Inspection Report 85-12 and an open question was remaining concerning the experience requirements as stated in licensee procedure TPP-7-024, "Shift Technical Advisor Training Program," which was not issued at the time. TPP-7-024, Revision 0, was issued March 8, 1985, endorsing INPO Guideline GPG-01, "Recommendations for Education and Training Nuclear Power Plant Shift Technical Advisors." This item is considered closed.

j. Item I.A.1.2, Shift Supervisor Responsibilities - This item requires assurances that the shift supervisor is not overburdened with administrative duties that could distract him from his principal responsibilities.

   Findings - This item was received in NRC Inspection Report 50-458/85-12 and remained open until all the positions of shift clerks were manned. On July 3, 1985, the last position for shift clerk was filled. This item is considered closed.

k. Item I.B.1.2, Independent Safety Engineering Group (ISEG) - This item is to establish an onsite independent safety engineering group (ISEG) to perform independent reviews of plant operations.

   Findings - This item was reviewed in NRC Inspection Report 50-458/85-24 and remained open until all positions in ISEG were manned. On June 10, 1985, the last position was filled bringing the total to five dedicated engineers. This item is considered closed.

6. Emergency and Abnormal Operating Procedures

The purpose of this area of the inspection was to ascertain whether or not plant emergency and abnormal operating procedures are prepared to adequately control safety-related functions in the event of system or component malfunction.

This inspection area was previously reviewed and left as an open item (8516-01) primarily because many of the procedures were being revised significantly and were not suitable for plant operations use. Except for the items noted below, the NRC inspector found that the concerns of open item (8516-01) have been resolved. Therefore, this item is considered closed.

The NRC inspector reviewed copies of the most recent revision to the licensee's Emergency Operating Procedures (EOPs) (Revision 2, issued April 23, 1985):

EOP-0001	"RPV Control"
EOP-0002	"Primary Containment Control"
EOP-0003	"Secondary Containment Control"
EOP-0004	"Level Restoration"
EOP-0005	"RPV Flooding"

The NRC inspector also reviewed copies of the following Abnormal Operating Procedures (AOPs):

AOP-0004	"Loss of Offsite Power," Rev. 1, 04-10-85
AOP-0010	"Loss of One RPS Bus," Rev. 1, 03-22-85
AOP-0013	"Loss of Containment Integrity," Rev. 1, 04-10-85
AOP-0015	"Loss of Drywell Cooling," Rev. 1, 04-18-85
AOP-0018	"Fuel Cladding Failure," Rev. 1, 05-03-85
AOP-0019	"Resin Intrusion to RCS," Rev. 1, 04-18-85
AOP-0021	"Anticipated Transient Without Scram," Rev. 0, 04-18-85
AOP-0030	"Control Rod Drive Malfunctions," Rev. 1, 04-10-85
AOP-0035	"SRV Stuck Open," Rev. 1, 05-03-85

Note: RPV - Reactor Pressure Vessel
RPS - Reactor Protection System
SRV - Safety Relief Valve
RCS - Reactor Coolant System

The NRC inspector identified the following problems with the EOPs and AOPs:

(1) Disagreement between the procedures and the Technical Specifications regarding alarm set points and values of parameters at which a Limiting Condition for Operation (LCO) becomes effective. Some examples are:

- Drywell High Pressure (The procedures say 2 psig instead of the correct valve of 1.68 psig of the Technical Specifications (TSs))

- Suppression Pool temperature at which a manual reactor scram is required in the event of a stuck open safety/relief valve. (The TS valve is 105°F, the procedure, AOP-0035, value is 110°F)

- Table 1 of EOP-0001, which is a list of alarm and trip set points, contains several set points that do not agree with the TSs.

(2) Throughout the EOPs and AOPs the licensee has provided a list of conditions, each of which are sufficient to require emergency RPV depressurization. Two conditions, listed separately in EOP-0002, are not included in the list, which states only six conditions. The list should be revised throughout to include all eight conditions.

(3) The scale on the fuel zone level instrumentation has been modified so that its zero reference point is the same as all other level instruments. The procedures have not been corrected to reflect this modification.

(4) Procedure AOP-0031 "Shutdown from Outside the Main Control Room," is not yet ready for plant operations use, apparently due to system modifications that are still ongoing.

(5) Procedure EOP-0002: Section 3.4, which deals with emergency venting of the primary containment is being rewritten in response to an NRC confirmatory item. (This item is being followed by NRR separately.)

The problems noted above, and any additional problems of a similar nature identified by the licensee, must be corrected prior to going beyond 5% of full power. This will remain as an open item (8552-01) until the NRC has verified these problems have been corrected.

7. Exit Meeting

The NRC inspectors conducted an exit meeting with the licensee personnel denoted in paragraph 1 on July 12, 1985. The NRC senior resident inspector also attended. As this meeting the findings of the inspection were summarized.