NUREG-0781
Supplement No. 1

Safety Evaluation Report
related to the operation of South Texas Project, Units 1 and 2

Docket Nos. 50-498 and 50-499

Houston Lighting and Power Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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ABSTRACT

In April 1986 the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-0781) regarding the application of Houston Lighting and Power Company (applicant and agent for the owners) for a license to operate South Texas Project, Units 1 and 2 (Docket Nos. 50-498 and 50-499). The facility is located in Matagorda County, Texas, west of the Colorado River, 8 miles north-northwest of the town of Matagorda and about 89 miles southwest of Houston. This first supplement to NUREG-0781 reports the status of certain items that remained unresolved at the time the Safety Evaluation Report was published.

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1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1. Introduction

In April 1986 the Nuclear Regulatory Commission (NRC) staff issued its Safety Evaluation Report (SER) (NUREG-0781) on the application filed by Houston Lighting and Power Company (HL&P), the applicant, acting on behalf of itself and the other owners [City Public Service Board of San Antonio (CPS), Central Power and Light Company (CPL), and City of Austin (COA)] for a license to operate South Texas Project, Units 1 and 2, Docket Nos. 50-498 and 50-499. At that time the staff identified items that had not been resolved with the applicant. The purpose of this supplement to the SER is to present the staff evaluation of open and confirmatory items that have been resolved, to report the status of unresolved items, to present the comments made by the Advisory Committee on Reactor Safeguards (ACRS) in its letter dated June 10, 1986, and to consider applicant's comments on the SER provided by letter dated June 9, 1986.

At its 314th meeting on June 5-7, 1986, the ACRS reviewed the application. In a June 10, 1986, letter from ACRS Chairman David A. Ward to NRC Chairman Nunzio J. Palladino, the Committee concluded that, subject to the resolution of open items identified by the NRC staff and the following three items noted in their letter:

(1) environmental qualification of the residual heat removal pump for operation inside containment in case of an accident

(2) resolution of Construction Appraisal Team (CAT) inspection findings

(3) testing and appropriate corrective measures to assure prevention of failures in the fuel oil piping and tubing by induced vibration resulting from extended operation of the diesel generators

there is reasonable assurance that the South Texas Project, Units 1 and 2, can be operated at power levels up to 3800 MWt without undue risk to the health and safety of the public.

The design of the plant was reviewed against Federal regulations, construction permit criteria, and the NRC "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP). Two versions of the SRP are in existence, NUREG-75/087 (March 1979) and NUREG-0800 (July 1981). Generally, the two versions contain very similar review criteria. Unless otherwise mentioned, reference to the SRP connotes the 1981 version.

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*Availability of all material cited is described on the inside front cover of this report.

Each of the following sections or appendices is numbered the same as the corresponding SER section or appendix that is being revised. Each section is supplementary to and not in lieu of the discussion in the Safety Evaluation Report unless otherwise noted. Appendix A continues the chronology of the staff's actions related to the processing of the South Texas Project application. Appendix B lists references cited in this report.* Appendix D contains abbreviations used in this supplement. Appendix E lists principal staff members and consultants who contributed to this supplement. Appendix K consists of a copy of the letter from the ACRS on South Texas Project, Units 1 and 2. Appendix L contains errata to the SER. Appendix M contains three documents: (1) the staff safety evaluation regarding tornado missile protection for the isolation valve cubicles previously transmitted to the applicant under cover letter dated January 6, 1984 (see Section 3.5.2), (2) a consultant's technical evaluation of probabilistic risk assessment for tornado and hurricane missile hazard to the containment isolation valve compartment equipment, and (3) a U.S. Department of Commerce letter dated December 24, 1983, evaluating supplemental information on the subject. Appendix N contains an EG&G Idaho, Inc. Technical Evaluation Report (TER) for South Texas Project conformance to Generic Letter 83-28 Items 2.1 (Part 1); Appendix O contains an EG&G Idaho, Inc. TER for conformance to Items 3.1.3 and 3.2.3 (see also Section 15.8.2).

Copies of this SER supplement are available for inspection at the NRC Public Document Room at 1717 H Street, N.W., Washington, D.C. and at the local Public Document Room located at the Wharton Junior College Library, Wharton, Texas.

The NRC Project Manager for South Texas Project, Units 1 and 2 is N. Prasad Kadambi. Dr. Kadambi may be contacted by calling (301) 492-7272. This supplement was prepared by NRC Project Manager Annette Vietti-Cook. Ms. Vietti-Cook may be contacted by calling (301) 492-8525. Both Project Managers can be contacted by writing to the Division of Licensing, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

1.7 Open Items

The staff identified certain open items in the SER that had not been resolved with the applicant. The status of these items is listed in an updated version of Table 1.4 and is discussed further in the sections of this report as indicated. If the staff has completed its review of an item, the notation "Resolved in SSER 1" so indicates. Supplement 1 resolves 4 open items, updates 2 open items (one of which is Generic Letter 83-28 for which many parts have been resolved), and adds 1 open item regarding vibration and wear on the bottom-mounted instrumentation thimbles inside the reactor vessel. The staff will complete its review of open items before the operating license is issued. Resolution of each of these open items will be discussed in future supplements to the SER.

1.8 Confirmatory Items

The staff identified confirmatory items in its SER that required additional information to confirm preliminary conclusions. The status of these items is listed in an updated version of Table 1.5 and is discussed further in the sections of this report as indicated.

If the staff has completed its review of an item, the notation "Resolved in SSER 1" so indicates. Supplement 1 resolves 5 confirmatory items, updates 3 confirmatory items (one of which is NUREG-0737 items for which many parts have been resolved), and adds 1 confirmatory item, "Compliance with Generic Letter 85-12 (TMI Item II.K.3.5)," which was separated from the other NUREG-0737 items.

1.9 License Condition Items

In Section 1.9 of the SER, the staff identified 3 license conditions. These are issues that must be resolved by the applicant as a condition for issuance of an operating license, and other longer term resolution issues that will be cited in the operating license issued, to ensure that NRC requirements are met during plant operation.

The current status of license conditions is in the updated version of Table 1.6.

Table 1.4 Listing of open items

Item	Status	SER Section
(1) Internal flooding analysis	Awaiting information	3.4.1, 9.2.7, 9.3.3
(2) Internal missiles analysis	Resolved in SSER 1	3.5.1, 10.4.9
(3) Staff review of jet impingement from high energy pipe failures	Resolved in SSER 1	3.6.1
(4) Equipment qualification		
(a) Seismic and dynamic qual.	Under review	3.10.1
(b) Pump and valve operability	Under review	3.10.2
(c) Environmental equipment qual.	Under review	3.11.3
(5) Preservice inspection/inservice inspection program review	Awaiting information	5.2.4, 6.6.1
(6) Design, verification, and validation of qualified display processing system	Awaiting information, under review	7.1.2
(7) Acceptability of isolation between safety and non-safety systems	Awaiting information	7.3.2.5
(8) Conformance to RG 1.97	Under review	7.5.2.4
(9) Test results of aluminum-sheathed and copper-sheathed cable	Awaiting information, under review	8.3.3.3
(10) Maximum available fault currents at electrical penetrations	Resolved in SSER 1	8.3.3.5
(11) Safe and alternate shutdown systems	Under review	9.5.1
(12) Auxiliary feedwater system reliability study	Resolved in SSER 1	10.4.9
(13) Emergency planning	Under review	13.3
(14) Industrial security	Under review, evaluation updated in SSER	13.6
(15) Analysis for boron dilution event during modes 4 and 5	Awaiting information	15.4.6
(16) Use of TREAT code for small-break loss-of-coolant-accident analysis	Awaiting information	15.6.5, 6.3.5

Table 1.4 (Continued)

Item	Status	SER Section
(17) Review of submittals on Generic Letter 83-28	Under review, evaluation updated in SSER 1	15.8.2
(18) Wear of the bottom mounted instrumentation thimbles	Awaiting information	3.9.2.3

Table 1.5 Listing of confirmatory items

Item	Status	SER Section
(1) Onsite meteorological measurements program (a) Comparison of new with old system (b) Measurement of precipitation to resolve and inconsistent records	Awaiting information	2.3.3
(2) Staff's independent analysis of the thermal performance of the essential cooling pond	Resolved in SSER 1	2.4.11.2
(3) Geotechnical monitoring program to detect horizontal and vertical movements	Resolved in SSER 1	2.5.1
(4) Review of stability and safety data relative to main cooling reservoir dike after filling to 49 feet msl	Awaiting information	2.5.7
(5) Completion of review of reports on pressure relief devices	Under review	3.9.3.2
(6) Design information on ASME Code Class 1, 2, and 3 component supports (Question 210.60)	Awaiting information	3.9.3.3
(7) Preservice and inservice testing of pumps and valves	Under review	3.9.6
(8) Acceptability of consequences from momentary liftoff of fuel assembly	Resolved in SSER 1	4.2.3.1(9)
(9) Combined seismic and loss-of-coolant accident loads on fuel assemblies	Awaiting information	4.2.3.3(4)
(10) Steam generator inspection	Awaiting information	5.4.2.2.2
(11) Applicability of Diablo Canyon natural circulation test	Under review	5.4.7
(12) Conservatism of loss-of-coolant accident analysis in light of information in FSAR Table 6.3.1 and response to Question 440.39	Resolved in SSER 1	6.3.1

Table 1.5 (Continued)

Item	Status	SER Section
(13) Analysis for nonisolable small-break loss-of-coolant accident	Combined with Open Item 16	6.3.5.2
(14) Reanalysis of loss-of-coolant accident during shutdown	Combined with Confirmatory Item 15	6.3.5.3
(15) Analyses for 6-inch and 8-inch breaks with justification for operator actions	Awaiting information	6.3.6, 15.6.5
(16) Interface between Class 1E and circuits	Under review	7.3.2.12
(17) Adequacy of design change so that main steam isolation valves do not operate on safety injection signal	Awaiting information	7.3.2.2
(18) Additional information on Criterion 2 of NUREG-0737 Item II.B.3	Awaiting information	9.5.2.2
(19) Procedures for preventive maintenance and operability checks on emergency communication equipment	Awaiting information	9.5.2.5
(20) Inservice inspection and testing of emergency dc lighting	Awaiting information	9.5.3
(21) Revision of radwaste process control program to meet staff guidelines	Awaiting information	11.4.2
(22) Update process and instrumentation diagrams for solid waste processing	Awaiting information	11.4.2
(23) Report on staff's site visit to corporate office and plant	Under review	13.1.1.3
(24) Conformance to Generic Letter 84-16 on hot operating experience	Under review	13.1.2.1
(25) Compliance with commitments on administrative procedures	Under review	13.5.1.9

Table 1.5 (Continued)

Item	Status	SER Section
(26) Qualification requirements for preoperational and initial startup test personnel to be equivalent to American National Standards Institute/American Nuclear Society Standard 3.1-1981	Resolved in SSER 1	13.5.1.9
(27) Staff review of the procedures generation package	Under review	13.5.2
(28) Plant-specific information on steam generator tube rupture	Under review	15.6.3
(29) Review of design against 10 CFR 50.62	Awaiting information, evaluation updated in SSER 1	15.8.1
(30) Results of the engineering assurance program	Awaiting information	17.4.3
(31) Results of the final verification and validation program for the final emergency operating procedures	Awaiting information	18
(32) Results of investigation of green Roto-tellite lights under actual operating conditions	Awaiting information	18
(33) Results of surveys of lighting, sound, meter, and communication system when control room work is completed	Awaiting information	18
(34) NUREG-0737 items:		
II.E.1.1 Auxiliary feedwater system evaluation	Combined with Open Item 12	10.4.9
II.E.3.1 Emergency power for pressurizer heaters	Resolved in SSER 1	8.3.6
II.G.1 Power supplies for pressurizer relief valves, block valves, and level indicators	Resolved in SSER 1	8.3.6

Table 1.5 (Continued)

Item	Status	SER Section
(34) NUREG-0737 items: (cont'd)		
II.K.1 IE Bulletins		
5. Review of ESF Valves	Resolved in SSER 1	6.3.1
10. Operability Status	Resolved in SSER 1	6.3.1
II.K.2 Orders B&W plants		
13. Thermal mechanical report: effect of HPI for small-break LOCA with no auxiliary feedwater	Resolved in SSER 1	15.6.5
II.K.3 Final recommendations, B&O task force		
3. Reporting SV and RV failures and challenges	Resolved in SSER 1	5.2.2.1
5. Automatic trip of RCPs	Combined with Confirmatory Item 35	15.6.5.1
17. ECCS outages	Resolved in SSER 1	6.3.1
25. Power on pump seals	Resolved in SSER 1	15.1.5.1
30. Small-break LOCA methods	Awaiting information	15.6.5
31. Compliance with 10 CFR 50.46	Awaiting information	15.6.5
III.A.1.2 Upgrading of emergency support facilities	Combined with Open Item 13	13.3
III.A.2 Emergency preparedness	Combined with Open Item 13	13.3
III.D.1.1 Primary coolant outside containment	Awaiting information	13.5
(35) Compliance with Generic Letter 85-12 (TMI Item II.K.3.5) RCP setpoint for small break LOCAs	Awaiting information, evaluation updated in SSER 1	15.6.5.1
1. Selected RCP trip criterion, including numerical values, final uncertainties, final calculated results, and a comparison of the criterion to the calculations which illustrates application of the uncertainties, and establishes the separation between LOCA and non-LOCA events.		

Table 1.5 (Continued)

Item	Status	SER Section
(35) Compliance with Generic Letter 85-12 (TMI Item II.K.3.5) RCP setpoint for small-break LOCAs (Continued) 2. Impact of an adverse containment atmosphere on RCS pressure indication as determined by the ongoing program. 3. Identification of the specific instrumentation used to determine the need for RCP trip. 4. Summary of the program to evaluate pipe whip and fluid jets insofar as impact upon RCP trip is concerned. 5. Definition of the radiation environment associated with an adverse containment. 6. Results of the program to evaluate pipe breaks insofar as it impacts upon RCP trip considerations.		

Table 1.6 Listing of license conditions

Item	Status	SER Section
(1) Implementation report on preoperational testing of the reactor vessel water level system	Awaiting information	4.4.6.3
(2) Postaccident qualification of the residual heat removal system before December 31, 1988, or the second refueling outage, whichever comes first	Awaiting information	5.4.7.7
(3) Implementation and maintenance in effective of all provisions of the approved fire protection program	Awaiting information	9.5.1.8

2 SITE CHARACTERISTICS

2.4 Hydrologic Engineering

2.4.11 Cooling Water Supply

2.4.11.2 Emergency Cooling Water

In the SER, the staff stated that it was conducting an independent evaluation of the thermal performance of the essential cooling pond (ECP) in accordance with Standard Review Plan (SRP) Section 2.4.11. The staff has now completed its evaluation using the criteria in Regulatory Guide (RG) 1.27, Revision 2, "Ultimate Heat Sink for Nuclear Power Plants."

The ultimate heat sink (USH) as defined in RG 1.27, consists of the main cooling reservoir (MCR) and the essential cooling pond. The ability of the MCR to provide cooling water for normal operation was discussed in Section 2.4.11.1 of the SER. Cooling water for postaccident shutdown and normal cooldown is provide by the ECP. Section 2.4.11.2 of the SER discusses the applicant's analysis of the performance of the ECP. The staff's independent analysis of the ECP is discussed below.

Emergency cooling water will be withdrawn from the essential cooling water (ECW) intake structure at an elevation of about 11 feet above mean sea level (MSL). Heated water will be discharged back to the ECP through the ECW discharge structure at an elevation of about 31 feet MSL. The configuration of the pond with withdrawal near the bottom and discharge near the surface will promote efficient cooling by stratifying water into hot and cool layers. In addition, the separation of the ECW intake and discharge structures, by a central dike in the ECP as shown in SER Figure 2.8, will ensure good circulation and prevent any short-circuiting between the intake and discharge.

Using the methods discussed in NUREG-0693, "Analysis of Ultimate Heat Sink Cooling Ponds," and NUREG-0733, "Analysis of Ultimate Heat Sink Spray Ponds," and the criteria of RG 1.27, the staff analyzed the performance of the ECP. Using long-term meteorological data from Victoria, Texas, the staff predicted that the highest temperature of water at the intake side of the ECP would be about 103.7°F. The adequacy of the meteorologic data base was determined by comparing the Victoria, Texas, data with 22 summer months of onsite data. This comparison showed that the Victoria, Texas, data were slightly more moderate and probably underestimated the ECP pond temperature by about 0.3°F. Adding 0.3°F to the highest pond temperature of 103.7°F results in a maximum temperature in the water withdrawn from the ECP of 104°F. This compares with a maximum temperature of 105°F calculated by the applicant.

Since the staff's analysis predicts a temperature slightly lower than the applicant's estimate, the staff concludes that the South Texas Project meets the guidelines of RG 1.27 and the requirements of General Design Criterion (GDC) 44 of Appendix A to 10 CFR 50.

2.4.14 Technical Specifications and Emergency Operating Requirements

In Section 2.4.14 of the SER, the staff stated that the plant will be permitted to operate only when the water level in the ECP is at or above elevation 25.5 feet MSL. The staff also stated that operation will be permitted only when the temperature in the ECP is less than a maximum value. The staff, however, did not specify what this maximum temperature would be. In performing its independent analysis of the thermal performance of the ECP, the staff assumed that the ECP water temperature on the intake side of the pond would be at 95°F at the start of the design-basis accident. This value was also used by the applicant in its analysis. On this basis, the staff concludes that plant shutdown should be initiated whenever the water level in the ECP drops to elevation 25.5 feet MSL or when the intake water temperature rises above 95°F. Thus an ultimate heat sink technical specification should define the actions to be taken in the event that the ECP water level drops below elevation 25.5 feet MSL or the water temperature at the intake side of the pond rises above 95°F.

2.5 Geology and Seismology

2.5.1 Basic Geologic and Seismic Information

2.5.1.2 Site Geology

In the SER, the staff stated it would review the commitment from the applicant of the detailed procedures to monitor horizontal and vertical movements periodically to ensure that there is no undetected subsidence, tilting, or differential movement on or in the plant site.

By letter dated July 7, 1986, the applicant stated that the data which are tabulated and charted in FSAR Table 2.5.C-3 and Figures 2.5.C-19, 2.5.C-19A, 2.5.C-23, 2.5.C-23A, 2.5.C-24, and 2.5.C-24A will be updated in the FSAR annually for the first 5 years after the Unit 1 operating permit is issued and that after the first 5 years, the data will be updated in the FSAR every 5 years. Further, the applicant stated that upon review of the data, should trends become apparent which may have a significant effect on the plant site, the NRC will be notified and immediate steps will be taken to update the FSAR tables and figures.

The staff considers this acceptable and, therefore, considers Confirmatory Item 3 closed.

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.5 Missile Protection

3.5.1 Missile Selection and Description

3.5.1.1/3.5.1.2 Internally Generated Missiles (Outside/Inside Containment)

In SER Sections 3.5.1.1 and 3.5.1.2, the staff stated that the applicant had not provided a missile generation analysis, postulating the failure of pumps, fans, pressurized tanks and compressed air/gas cylinders, which shows that safe shutdown will not be affected, even after considering the single active failure criterion.

The applicant, by letter dated June 17, 1986, provided the results of its missile generation analysis for both inside and outside containment based on postulated failures of pumps, fans, pressurized tanks, and compressed air/gas cylinders. The applicant confirmed that potential missiles from these sources will not prevent safe shutdown of the plant or result in an uncontrolled release of radioactivity, considering single active failures. Barriers are provided for missile protection which are oriented to either contain postulated missiles or deflect them from essential (safety-related) equipment. Compressed air/gas cylinders are, for example, separated from safety-related components by placing them in cubicles or subcompartments within structures, or are restrained. The staff finds this approach acceptable.

In SER Sections 3.5.1.1 and 10.4.9, the staff required the applicant to confirm that potential missiles from the auxiliary and main feedwater pump turbines, regardless of redundant overspeed trip capability, will not damage other essential equipment. By letter dated June 17, 1986, the applicant confirmed that a review of the potential missiles from pump turbines indicates that no essential equipment will be adversely affected. Missiles arising from a postulated failure of an auxiliary feedwater pump as a result of 120% overspeed will not have enough energy to penetrate the turbine housing. In addition, the trains of main steam, main feedwater, and auxiliary feedwater equipment in an isolation valve cubicle are separated from each other by 2 feet of concrete. Potential missiles from the main feedwater pump turbines in the turbine building will not have sufficient energy to penetrate the walls of seismic Category I structures where essential equipment is located. The staff finds this rationale acceptable.

On the basis of the above, the staff concludes that the applicant has provided adequate protection for safe-shutdown equipment against internally generated missiles, both inside and outside containment, and that the provisions of General Design Criterion (GDC) 4 of Appendix A to 10 CFR 50 and staff guidelines provided in Standard Review Plan (SRP) Sections 3.5.1.1 and 3.5.1.2 have been satisfied.

3.5.2 Structure, Systems, and Components To Be Protected From Externally Generated Missiles

In the SER, the staff identified that the applicant, during the design of the isolation valve cubicles (IVCs) for missile protection, elected to demonstrate compliance with the tornado missile protection criteria in SRP Sections 3.5.1.4 and 3.5.2 by the use of probabilistic risk assessment rather than by providing positive protection for these cubicles by such means as a missile-proof roof.

By letter dated January 6, 1984, the staff transmitted its evaluation on tornado missile protection for the IVCs. As indicated in the SER, the staff concludes that the probability of tornado missile damage to the IVCs and associated essential equipment has been adequately demonstrated to be approximately 3 x 10^{-9} per year. Further, the thick concrete walls which constitute the sides of the IVCs provide protection for the equipment contained therein from all tornado missiles except those entering the cubicles through the open roof area which the staff considers to be a low probability event. Thus, the applicant has satisfactorily demonstrated compliance with the requirements of GDC 2 and 4 with respect to tornado missile protection for the IVCs, and the safety-related equipment within them.

For completness, the staff's safety evaluation and consultants' technical evaluation report transmitted in the January 6, 1984, letter are provided in Appendix M.

3.6 Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

Section 3.6.1 of the SER states that review of the applicant's analysis of postulated high- and moderate-energy pipe breaks outside containment (including pipe whip, jet impingement, flooding, and environmental effects on safety-related equipment) is not complete. The staff has now reviewed the high-energy pipe break analysis for the steam and feedwater piping located in the four isolation valve cubicles. This piping is in the main steam and feedwater break exclusion zones (i.e., superpipe areas). The applicant postulated a full circumferential (single area) break in these lines to ensure that the environmental and subcompartment pressurization effects, including superheat, will not result in unacceptable consequences. The applicant has also analyzed for jet impingement effects from branch lines in the superpipe area. However, the applicant did not perform an analysis of jet impingement effects in the break exclusion zone which is a deviation from SRP Section 3.6.1 of NUREG-0800.

The applicant's analysis is more conservative with respect to environmental effects than the 1-ft^2 break prescribed in SRP Section 3.6.1 of NUREG-0800. Also, essential equipment is protected against jet impingement and pipewhip effects from postulated full circumferential breaks of branch piping in the break exclusion area of the main steam or feedwater lines. The above design conforms to the guidelines of Branch Technical Position (BTP) APCSB 3-1, attached to SRP Section 3.6.1 of NUREG-75/087.

The staff has evaluated the South Texas Project main steam and feedwater break exclusion zone and considers the above deviation from NUREG-0800 acceptable. The basis for staff acceptance of this deviation is that NUREG-0800 prescribes jet impingement analyses as a means to achieve separation even though SRP Section 3.6.2 (NUREG-0800) states that breaks need not be postulated in break exclusion zone piping. Also, essential structures and equipment, are conservatively designed to be protected from, or qualified to withstand, the environmental and pressurization effects resulting from a nonmechanistic, full circumferential break in main steam or main feedwater piping and jet impingement and pipewhip effects resulting from postulated breaks in branch piping. Furthermore, the South Texas Project design separates the four main steamlines from each other and from essential equipment not affiliated with an individual steamline. This separation meets the objective of SRP Section 3.6.1 of NUREG-0800 and NUREG-075/87. On the above basis, the staff concludes that protection against pipe breaks in the break exclusion zone for South Texas Project, Units 1 and 2, meets the requirements of GDC 4, "Environmental and Missile Design Bases," with respect to protection from the environmental, jet impingement, and pipewhip effects from postulated pipe breaks.

The applicant's analysis of postulated high- and moderate-energy pipe breaks for locations outside the break exclusion zone of the main steam and feedwater piping, including flooding and environmental effects on safety-related equipment and structures, is not complete. Pending the receipt of this analysis, the staff cannot conclude that the applicant has adequately designed and protected structures, systems, and components required for safe plant shutdown following postulated pipe break events, i.e., the combined pipe break and single active failure.

On the basis of the foregoing discussion, the staff concludes that the South Texas Project design for protection against piping failures outside containment meets the requirements of GDC 4 in the break exclusion area of the main steam and feedwater lines, despite the differences in the analytical prescriptions in the NUREG-75/087 and NUREG-0800 versions of SRP Section 3.6.1. The staff considers Open Item 3, staff review of jet impingement from high energy pipe failures, resolved and will address the applicant's analysis of postulated high- and moderate-energy pipe breaks outside containment as it relates to flooding effects on safety-related equipment as part of Open Item 1, internal flooding analysis (SER Sections 3.4.1, 9.2.7, and 9.3.3), in a future supplement. Further, the applicant's analysis of postulated high- and moderate-energy pipe breaks outside containment as it relates to environmental effects on safety-related equipment will be addressed as part of Open Item 4(c), environmental equipment qualification (SER Section 3.11.3), in a future supplement.

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping

In Section 3.6.2 of the SER, the staff based its conclusions relative to the dynamic effects of jet impingement loads from a postulated pipe break on information in Section 3.6.2 of the FSAR which was compared with the criteria in SRP Section 3.6.2. In a letter from M. R. Wisenburg to V. S. Noonan, "Revision to FSAR Section 3.6 - Two Phase Jet Criteria," dated May 2, 1986, the applicant submitted screening criteria relative to the distances over which jet impingement loads are effective. These criteria are not totally consistent with SRP Section 3.6.2. Specifically, the applicant's screening criteria are that components impacted by jets from postulated breaks in piping containing high-pressure (870 to 2465 psia) steam or subcooled liquid that flashes at the break shall be evaluated as follows:

(1) Unprotected components within a distance of 10 pipe diameters from the broken pipe are assumed to fail. Specific jet loads are calculated and evaluated when failure of the component, combined with a single active failure could adversely affect safe-shutdown capability. These jet load calculations will be performed in accordance with FSAR Section 3.6.3.3.1, which is consistent with the guidelines in SRP Section 3.6.2.

(2) Unprotected components at a distance beyond 10 pipe diameters from the broken pipe are considered undamaged by the jet without further analysis.

The assumption in screening criterion 1 (above), that all components within 10 diameters of the break will fail, is more conservative than the staff's guidelines. The remainder of criterion 1 is consistent with the staff's position. On the basis of the above information, the staff has concluded that screening criterion 1 is acceptable.

The basis for screening criterion 2 (above) is contained in NUREG/CR-2913, "Two Phase Jet Loads," dated January 1983. The staff has reviewed NUREG/CR-2913 and has concluded that the methodology therein which predicts that jet impingement loads are negligible at a distance beyond 10 pipe diameters from breaks in pipes containing high-pressure steam or subcooled liquid that flashes at the breaks is acceptable. On the basis of the above information, the staff has concluded that screening criterion 2 is acceptable.

3.9 Mechanical Systems and Components

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

3.9.2.3 Preoperational Flow-Induced Vibration Testing of Reactor Internals

In Section 3.9.2.3 of the SER, the staff discussed the relevance of a wear problem of the bottom-mounted instrumentation (BMI) thimbles in the Paluel (French) plant to the South Texas plant. On the basis of information available at the time the SER was written, which included a limited amount of operating experience from two Belgian plants (DOEL-4 and Tihange-3) with BMI thimbles similar to those at South Texas, the staff concluded that the South Texas thimbles will perform satisfactorily.

In a letter dated June 27, 1986, the applicant reported that further operating experience at DOEL-4 and Tihange-3 has shown that the thimbles on both plants are experiencing accelerated wear. The applicant also reported that a design study is under way with Westinghouse to develop an appropriate modification for these thimbles for South Texas, Units 1 and 2, and the staff will be kept informed about the evaluations.

Since a failure of these thimbles could result in a small break in the reactor coolant pressure boundary, which apparently cannot be isolated, the staff is identifying this as Open Item 18. The results of the staff's evaluation of this issue will be presented in a supplement to the SER.

4 REACTOR

4.2 Fuel Design

4.2.3 Design Evaluation

4.2.3.1 Fuel System Damage Evaluation

4.2.3.1(9) Assembly Liftoff

In the SER, the staff stated that the applicant should confirm that no adverse consequences of momentary liftoff were expected during a turbine overspeed transient. In a letter from M. R. Wisenburg (HL&P) to V. S. Noonan (NRC), dated July 25, 1986, the applicant showed that the amount of assembly momentary liftoff is small; i.e. the whole assembly still remains seated in the lower core plate during the turbine overspeed transient. Therefore, the staff concludes that the assembly liftoff will have no adverse effects on adjacent assemblies during the turbine overspeed transient and Confirmatory Item 8 is resolved.

5 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.2 Overpressure Protection

5.2.2.1 Overpressure Protection During Power Operation

In response to the requirement of NUREG-0737 Action Item II.K.3.3, "Reporting SV and PORV Failures and Challenges," the applicant stated that the South Texas Project Technical Specifications will include a requirement to promptly report to NRC a failure of a power-operated relief valve (PORV) or a safety valve to close and will also include a requirement to document challenges to the PORVs or safety valves in the monthly operating report. The staff will review the South Texas Project Technical Specifications for compliance with this commitment.

6 ENGINEERED SAFETY FEATURES

6.3 Emergency Core Cooling System

6.3.1 System Design

In the SER, the staff noted that a discrepancy existed with regard to the value of the high-head safety-injection (HHSI) pump shutoff head, listed as 3900 feet in FSAR Table 6.3-1, but, in response to staff question 440.39, stated to be 1445 psig (about 3400 ft). The staff requested that the applicant confirm that the loss-of-coolant accident (LOCA) analyses were performed using the more conservative value. The applicant's response indicated that the HHSI pump design shutoff head is 3900 feet, but that the South Texas LOCA analyses minimum safeguards assumptions include the effects of a spilling line, conservative line resistance values, and degraded pump head-flow parameters, with a consequent conservative shutoff head of 1445 psig. The staff concludes that the applicant's response is satisfactory and that this issue is resolved.

In response to the requirement of NUREG-0737 Action Item II.K.1.5, "Review of Safety-Related Valve Positions, Controls, and Related Test and Maintenance Procedures To Assure Proper Functioning," the applicant has stated that plant procedures provide the necessary verifications to ensure that valves are maintained in their correct positions during all operational modes. Safety-related valve positions, positioning requirements, and controls have been reviewed to ensure that valves remain in their correct positions for engineered safety feature (ESF) operations. On this basis, the staff concludes that the applicant's commitment meets the guidelines of this item and is acceptable.

In response to the requirement of NUREG-0737 Action Item II.K.1.10, "Review and Modification of Maintenance and Test Procedures for Removal of Safety-Related Systems From Service and Verification of the Operability of Safety-Related Systems When They Are Returned to Service," the applicant stated that plant procedures require verification that redundant safety-related components are available before any safety-related component is removed from service. Plant procedures require verification of the operability of safety-related systems when they are returned to service following maintenance or testing. Plant procedures also require notification of appropriate operational personnel when a safety-related system is removed from or returned to service. On this basis, the staff concludes that the applicant's commitment meets the guidelines of this item and is acceptable.

In response to the requirement of NUREG-0737 Action Item II.K.3.17, "Report on Outages of Emergency Core Cooling Systems (ECCS)," the applicant in FSAR Appendix 7A committed to report ECCS outage data to the NRC. The staff finds this acceptable and will review the South Texas Technical Specifications for compliance with this commitment.

8 ELECTRICAL POWER

8.3 Onsite Power System

8.3.3 Compliance With GDC

8.3.3.5 Compliance With GDC 50

The staff stated in the SER that the coordinated primary and backup protection curves submitted by the applicant for the circuits passing through electrical penetrations did not show the maximum available fault current at the penetrations to demonstrate that adequate time-current coordination exists between the primary and backup protection devices and the penetration itself.

By letter dated June 11, 1986, the applicant submitted the revised time-current characteristic curves of protective devices showing the maximum available fault current at the penetration for each size of penetration for staff review. On the basis of its review, the staff concludes that the South Texas design provides independent primary and backup fault protection, for each penetration conductor, to preclude a single failure from impairing the integrity of a containment electrical penetration. This meets the requirements of GDC 50 and the guidance of Regulatory Guide 1.63 and is, therefore, acceptable.

8.3.6 Compliance With the Guidelines of NUREG-0737 Action Items II.E.3.1 and II.G.1

NUREG-0737 Action Item II.E.3.1, "Pressurizer Heater Power Supply," requires that emergency power be available to a minimum number of pressurizer heaters to maintain hot standby conditions when offsite power is lost. The South Texas design provides two Class 1E pressurizer heater groups supplied from separate Class 1E systems, one from engineered safety feature (ESF) Train A and one from ESF Train C. These buses are energized from separate and independent diesel generators upon loss of offsite power. The connection of the pressurizer heaters and controls to the Class 1E buses is through safety-grade circuit breakers. Procedures for manually loading the pressurizer heaters onto Class 1E buses following a loss of offsite power will be available to the operator.

On the basis of its review, the staff concludes that the power supply criteria used in South Texas plant for pressurizer heaters are consistent with the guidelines of NUREG-0737 and, therefore, are acceptable.

NUREG-0737 Action Item II.G.1, "Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators," requires that the power supply and associated controls to pressurizer equipment be safety grade and that the power supply shall be capable of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

Two parallel sets of power-operated relief valves (PORV) and PORV block valves are provided: one set is assigned to Train A and the other set is assigned to Train B. The PORVs for the South Texas plant are powered from the Class 1E dc system. The power supplies to the block valves are from two redundant 480-V ac buses that are powered automatically from their respective diesel generators on loss of offsite power.

The pressurizer level instrumentations are Class 1E and are powered from Class 1E buses which are capable of being powered from the diesel generators on loss of offsite power.

On the basis of its review, the staff concludes that the power supplies and associated controls for PORVs, block valves, and pressurizer level instruments are safety grade and the power supplies are capable of being powered from both offsite power and the onsite emergency power systems. This is consistent with NUREG-0737 requirements and, therefore, is acceptable.

10 STEAM AND POWER COVERSION SYSTEM

10.4 Other Features

10.4.9 Auxiliary Feedwater System

In Section 10.4.9 of the SER, the staff required further information to confirm compliance with General Design Criterion (GDC) 4 (Appendix A to 10 CFR 50) as it relates to protection against internally generated missiles, GDC 19, and Branch Technical Position (BTP) RSB 5-1 regarding the capability to achieve cold shutdown from the control room using only safety-related equipment. (Refer to Sections 3.5.1.1 and 3.5.1.2, "Internally Generated Missiles," for closure of open items on internally generated missiles.)

In Section 10.4.9 of the SER, the staff indicated that the applicant's auxiliary feedwater system (AFWS) reliability study was under review and that the results of the review would be provided in a supplement to the SER. The staff has now completed its review, and its evaluation of the AFWS reliability study follows.

The applicant's AFWS reliability study was provided in accordance with NUREG-0737 Action Item II.E.1.1 and appears in Appendix 10A of the South Texas Project, Units 1 and 2, FSAR. The original submittal, dated October 21, 1985, contained the results of an AFWS reliability analysis based on the conservative assumption that 3 out of 4 auxiliary feedwater (AFW) trains were operable at the inception of a transient. In a letter dated March 11, 1986, the applicant submitted a revised AFWS reliability study which removed the aforementioned conservatism regarding the unavailability of the fourth AFW train; the fourth train is now considered to be operable. However, its availability is reduced by assuming much longer outage times for maintenance; the evaluation is now done on the basis of a continuous, 2-week outage for the fourth train compared to 19-hour outage times for each of the remaining three trains. The staff and its consultant, the Brookhaven National Laboratory (BNL), have reviewed the AFWS reliability study relative to system unavailability in providing sufficient water to the steam generators for decay-heat removal in the event of loss of main feedwater (LMFW), loss of offsite power (LOOP), or loss of all ac power (LOAC). In addition, BNL has performed an independent AFWS reliability analysis. The BNL results, compared with those obtained by the applicant, are presented below:

Case I - Tabulation of AFWS unavailability on demand for various transients, based on applicant's original submittal which assumed 3 out of 4 trains operable.

Transient	Applicant results	BNL results
LMFW	1.82×10^{-5}	3.90×10^{-5}
LOOP	3.96×10^{-5}	2.10×10^{-4}
LOAC	5.90×10^{-2}	5.94×10^{-2}

The difference in the values obtained for the LOOP transient is primarily due to the assumed unavailability of both diesel generators (motor-driven AFW pumps) in the BNL analysis. That is, one diesel generator is assumed to be out of service for maintenance and the other is incapacitated because of a single active failure. The value obtained by the applicant was based on the assumption that one diesel generator (motor-driven AFW pump) was always available (availability of 1.0), and the motor- and turbine-driven AFW pumps were unavailable as a result of single active failure and maintenance considerations.

Case II - Tabulation of AFWS unavailability on demand for various transients, based on applicant's revised analysis which assumed all 4 trains operable but a much larger unavailability of the fourth (motor-driven AFW pump) train because of maintenance.

Transient	Applicant results	BNL results
LMFW	3.23×10^{-6}	5.02×10^{-6}
LOOP	3.57×10^{-5}	4.88×10^{-5}
LOAC	4.54×10^{-2}	4.64×10^{-2}

For the LMFW and LOOP transients, a comparison of the above two tables indicates that if all four AFW trains are assumed operable, AFWS unavailability is reduced. Since this approach is acceptable, the results shown for Case II are considered to be reasonable estimates of the South Texas Project AFWS unavailabilities.

The staff has reviewed the BNL analysis and concurs with its findings. Therefore, on the basis of the foregoing discussion, the staff concludes that AFWS unavailability per demand for the South Texas Project is in the acceptable range (10^{-4} to 10^{-5}) for LMFW and LOOP transients (see SRP Section 10.4.9 of NUREG-0800). For the LOAC case, there is no unavailability criterion prescribed in staff guidelines; the results of the study do, however, ensure AFWS availability independent of ac power. Therefore, the staff concludes that the applicant has complied with the guidelines of NUREG-0737 Action Item II.E.1.1 concerning the AFWS reliability analysis and that AFWS unavailability per demand for the South Texas Project is compatible with staff guidance in SRP Section 10.4.9 (NUREG-0800).

With the above information, the staff considers Open Item 12 closed. The staff will continue to carry License Condition 2, "Postaccident qualifications of the residual heat removal system before December 31, 1988, or the second refueling outage, whichever comes first."

13 CONDUCT OF OPERATIONS

13.1 Organizational Structural of Applicant

13.1.2 Operating Organization

13.1.2.1 Plant Organization

13.1.2.1(1) Reactor Operations Division

In the SER, the staff stated that the applicant's plan to provide engineering expertise on shift by the use of shift technical advisors (STAs) is acceptable. Further, the staff indicated that the Commission, in Generic Letter 86-04, "Policy Statement on Engineering Expertise on Shift," issued on February 13, 1986, requests licensees and applicants to review their programs for providing engineering expertise on shift and to advise the Commission of any changes they propose to make to take advantage of the options identified in the Commission's Policy Statement. Option 1 provides for elimination of the separate STA position by allowing licensees to combine one of the required Senior Reactor Operator (SRO) positions with the STA position into a dual-role position (SRO/STA). Option 2 states that a licensee may continue to use an NRC-approved STA program while meeting licensed operator staffing requirements.

In response to the generic letter, the applicant, by letter dated May 8, 1986, stated that it will submit, in a forthcoming FSAR amendment, changes to FSAR Appendix 7A, Section I.A.1.1 which would authorize the use of either Option 1 (SRO/STA dual-role position) or Option 2 (STA position) to meet the engineering-expertise-on-shift requirements. This would allow the use of Option 1 on one or more shifts given that personnel qualify to fill the dual role position in the future. The staff finds this change acceptable.

13.5 Plant Procedures

13.5.1 Administrative Procedures

13.5.1.8 Conduct of Initial Test Program

In the SER, the staff stated that the applicant's commitment to Regulatory Guide (RG) 1.58 is to be improved for those persons in responsible positions for the initial startup testing to be equivalent to ANSI/ANS 3.1-1981. By letter dated April 14, 1986, the applicant provided additional information regarding the qualifications of personnel performing the initial startup testing, supplementing the information provided in Section 14.2.2.8 of the FSAR. The staff has reviewed this information and has determined that the qualifications of these personnel are sufficient to satisfy the criteria of Section 4.4.6 of ANSI/ANS 3.1-1981. Accordingly, the staff concludes that the qualifications of the personnel performing initial startup testing are acceptable.

13.5.1.9 Summary and Conclusions

The applicant has described the program and procedures that provide administrative controls over activities important to safety, including the control of the initial test program. The applicant meets Standard Review Plan (SRP) Section 13.5.1, except for the following Confirmatory Item 25:

> The staff will verify the applicant's compliance with commitments to have administrative procedures to control shift supervisor responsibilities,control room access, working hours, shift relief and turnover, feedback of operating experience, verification of operating activities, and crane operations.

13.6 Industrial Security

The applicant has submitted documents entitled "South Texas Project Electrical Generating Station Security Plan," "South Texas Electrical Generating Station Security Personnel Training & Qualification Plan," and "South Texas Project Electrical Generating Station Safeguards Contingency Plan" for protection against radiological sabotage. The plans were reviewed in accordance with SRP Section 13.6, "Physical Security."

As a result of the staff's evaluation, certain portions of these plans have been identified as requiring additional information and upgrading to satisfy the requirements of 10 CFR 73.55 and Appendix B of 10 CFR 73. Accordingly, the Security Plan and the Guard Training and Qualification Plans for the South Texas plant remain as Open Item 14. The Safeguards Contingency Plan has been approved and requires no additional modification.

The applicant's security plans are being protected from unauthorized disclosure in accordance with 10 CFR 73.21.

15 ACCIDENT ANALYSIS

15.1 Increase in Heat Removal by the Secondary System

15.1.5 Steam System Piping Failures Inside and Outside Containment

15.1.5.1 Steamline Rupture

NUREG-0737 Item II.K.3.25

In response to the requirement of NUREG-0737 Action Item II.K.3.25, "Effect of Loss of AC Power on Pump Seals," the applicant indicated that in the event of loss of offsite power, the reactor coolant pump (RCP) motors are deenergized and both seal injection flow from the chemical and volume control system (CVCS) and component cooling water (CCW) flow to the thermal barrier heat exchangers are temporarily terminated. However, the diesel generators are automatically started and either seal injection flow or CCW to the thermal barrier heat exchanger is restored by loading the charging and CCW pumps to the diesel generators. In accordance with FSAR Table 8.3-5, the charging pumps are loaded to the diesel generators in 10 seconds and the CCW pumps in 20 seconds after the diesel generator breakers are closed.

On this basis, the staff concludes that the applicant's design meets the requirements of Item II.K.3.25 and is acceptable.

15.6 Decrease in Reactor Coolant Inventory

15.6.5 Loss-of-Coolant Accident Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

15.6.5.1 Loss-of-Coolant Accident

NUREG-0737 Item II.K.2.13

The staff review of NUREG-0737 Action Item II.K.2.13, "Thermal Mechanical Report - Effect of High Pressure Injection on Vessel Integrity for Small-Break LOCA With No Auxiliary Feedwater," is addressed in conjunction with Unresolved Safety Issue A-49, "Pressurized Thermal Shock," in Appendix C of the SER.

The material that follows is self-contained, covers material provided by the applicant since the staff's previous SER was written, and should replace material in the previous SER pertinent to reactor coolant pump (RCP) trip.

Generic Letter 85-12

Generic Letter (GL) 85-12, "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps'," required owners of Westinghouse nuclear steam generating systems to evaluate their plants with respect to RCP trip. The objective was to demonstrate that their proposed RCP trip setpoints assure pump trip for small-break LOCAs, and, in addition, to provide reasonable assurance that RCPs are not tripped unnecessarily during non-LOCA events. A number of plant-specific items were identified which were to be considered by applicants and licensees, including the selected RCP trip parameter, instrumentation quality and redundancy, instrumentation uncertainty, possible adverse environments, calculational uncertainty, potential RCP and RCP-associated problems, operator training, and operating procedures.

The applicant has addressed each of the criteria in letters dated November 6, 1985, January 28, 1986, and May 30, 1986, and the staff has evaluated this information.

Organization of the material which follows meets the intent of GL 85-12 and appendices to which the applicant responded. A statement is first presented which describes the generic letter request. This is followed by a staff summary of the applicant's position and a staff evaluation.

Section A: Determination of RCP Trip Criteria

- Demonstrate and justify that proposed RCP-trip setpoints are adequate for small-break loss-of-coolant accidents (LOCAs) but will not cause RCP trip for other non-LOCA transients and accidents such as steam generator tube ruptures (SGTRs). This is to include performance of safety analyses to prove the adequacy of the setpoints.

- Consider using partial or staggered RCP-trip schemes.

The applicant is tentatively planning to use reactor coolant system (RCS) pressure as the criterion for RCP trip, but the final decision has not been reached. An alternate is the pressure difference between the RCS and the steam generator (SG) secondary side. A number of factors are under consideration pertinent to the final selection. For example, the pressure difference is displayed directly in the control room so that operators do not have to perform a subtraction in the process of reaching a trip/no-trip decision. However, the display position is not as attractive as is the case for RCS pressure. A human factors evaluation remains to be completed before selection. Further information pertinent to the selection is provided in the Subsection A2 response (see below). The staff finds this approach acceptable for the reasons discussed in Subsection A2.

Westinghouse does not support keeping the RCPs running if they should have been tripped but were not tripped before entering a window in time during which tripping RCPs will make the accident worse. The applicant's position is that the RCPs should be tripped at the time the error is discovered regardless of RCS conditions, unless one is obviously in an inadequate core cooling situation. The applicant argues, that if RCPs should be running under LOCA conditions, this will be the result of following the emergency operating procedures, since RCPs will be restarted in accordance with those procedures if such action is necessary. Reference was made to report number WOG-117 from the Westinghouse Owners Group (WOG) which addresses this situation. It was pointed out that the safety evaluation presented in GL 85-12 did not disallow this approach, and that it referenced WOG-117 results. The technical justification for the decision is that Westinghouse best estimate (BE) calculations show that the maximum cladding temperature never exceeds 2200°F, but always remains significantly below that value.

The staff has previously stated that RCPs are to be left running if they are not tripped in time to avoid the "window." Typical concerns are that there is no assurance the RCPs can be restarted under abnormal conditions if they are stopped; it has not been established that the liquid and vapor phases can be rehomogenized if the RCPs are stopped so that the phases separate, and stopping RCPs under "window" conditions can result in immediate creation of severe core heating conditions where such did not occur as long as the RCPs were running.

The staff recognizes that the applicant, as well as a number of other licensees, incorporated the Westinghouse recommendations into emergency operating procedures. The situation under consideration here is highly unlikely, involves conditions beyond the plant design basis, and probably can be handled by following either the staff or the applicant positions. Therefore, the staff will accept the applicant's position for now. The staff will address this item on a generic basis at a future time.

There are additional items of a similar nature which the staff will address at a future time. For example, many applicants and licensees are following guidance wherein they do not trip RCPs if no safety injection (SI) or charging is available. Again, this is beyond the design basis of the plants, involves a situation that was not covered in the original staff evaluation of RCP trip, is generic in nature, and is an unlikely event. Such items will be addressed generically in the future.

Subsection A1

- Identify the instrumentation to be used to determine the RCP trip setpoint, including the degree of redundancy of each parameter signal needed for the criterion chosen. Establish the quality level for the instrumentation, identify the basis for the sensing-instruments' design features, and identify the basis for the degree of redundancy.

RCS pressure will be determined by using three wide-range pressure measurements. The sensing lines are located on three of the four hot legs. The transmitters are located outside containment, and hence are not subject to an adverse containment atmosphere. They are Class 1E safety-related, and are fully qualified. There is a program in place, yet to be completed, to evaluate the influence of an adverse containment atmosphere on the pressure readings as seen by the operator.

These pressure-sensing devices were previously evaluated for response time with respect to a number of control situations. The only one encountered where response time was inadequate was for use in control of a cold overpressure condition. For that situation, the South Texas plant uses other connections to the RCS for which the transmitters are located within containment. These alternate pressures are also available to the operators as a source of further information, should it be needed for RCP trip.

The steamlines from each steam generator are equipped with three channels of pressure instrumentation which provide both monitoring and protection functions. These are fully qualified instrumentation packages. The transmitters are located in separated cubicles, one for each of the four steam generators. Hence, only one pressure reading would be expected to be affected by a steamline break in the vicinity of a transmitter.

The applicant has identified the instrumentation as fully qualified and has reported results from environmental considerations. The applicant has investigated many aspects of interaction of the environment with the instrumentation, and has stated that the remainder are to be evaluated via a program that is in place. The potential for adverse impact upon individual channels of instrumentation has been considered, and the means for dealing with loss of individual channels has been described. The staff agrees with the applicant's assessment that containment considerations probably will have little influence on pressure readings since the transmitters are outside containment. The specific instrumentation should be identified when the RCP trip criterion has been determined.

Response time, although not identified as an item of concern in the original staff review, could be a consideration under transient conditions for instrumentation which depends upon long, small-diameter pipe runs to transmitters. The applicant considered this earlier in studies in which the only unsatisfactory condition was found to be one corresponding to extremely rapid pressure responses involving relatively small changes in pressure. The requirements for RCP trip are far removed from that condition, and no difficulty is anticipated from long pipe runs.

The availability of backup instrumentation has also been considered in selection of instrumentation. The staff observes that the applicant is paying attention to human factors considerations.

The staff finds the items identified in Subsection A1 acceptable, subject to the clarifications identified in the previous paragraphs.

Subsection A2

- Identify the instrumentation uncertainties for both normal and adverse containment conditions. Describe the basis for the selection of the adverse containment parameters. Address, as appropriate, local conditions, such as fluid jets or pipe whip, which might influence the reliability of instrumentation.

The basis for selecting an adverse containment environment is identified as the WOG Emergency Response Guidelines (ERGs).

Instrumentation inaccuracy and uncertainty investigations have been completed, and the applicant has determined that either RCS pressure or RCS pressure to SG secondary side pressure differential is satisfactory as an RCP trip indicator. Subcooling margin did not provide satisfactory separation of LOCA and SGTR.

The limiting event for RCS pressure has been determined to be a feedline break, and the analysis for this application has not been completed. The limiting event for pressure difference is SGTR.

The uncertainty associated with RCS pressure is approximately 80 psi, and the expected uncertainty for pressure difference is approximately 70 to 90 psi. Further information will be provided to the staff regarding treatment of sensing line uncertainty and the difference between adverse and normal environmental conditions.

In the case of RCS pressure, the display processing takes an average of the three readings, analyzes the deviations from the average, and then, if one of the readings is outside of the established range, it is rejected, and a new average is determined. This should reduce operator involvement with situations where one instrument is faulty. Pressure information for each steam generator is treated in the same manner.

There is an ongoing program of evaluation of pipewhip and the influence of local fluid jets. This is not complete. Information will be provided to the staff when it becomes available.

An adverse containment environment is considered to correspond to a Hi-1 containment isolation signal, which is received at a pressure of 5.5 psig. The radiation environment associated with an adverse containment condition, if any, has not been identified. The applicant should address this item.

The information that has been provided and information that is anticipated should be sufficient to establish the adequacy of either of the two techniques that the applicant is considering.

The determination of instrumentation uncertainties has been described to the staff, and approximate values have been quoted that are within the range the staff would expect. The basis for determination of an adverse containment condition has been identified. Substantial work appears to have been completed on the potential impact of local conditions, and other work has been described as ongoing. Sufficient information has been reviewed that the staff does not expect any significant change in what has been reported, and the uncertainties associated with either RCP trip criterion are expected to be acceptable.

Operator response is under consideration, and some of the procedures the operator will follow have been developed in anticipation of selection of a trip criterion. Other responses have been described and, in general, the WOG emergency procedures are being followed. This deals with many aspects of RCP operation.

The above information leads the staff to conclude that the topics identified in this subsection are satisfactorily covered by the applicant, subject to the confirmatory items which have been identified.

Subsection A3

- In addressing criterion selection, provide consideration of uncertainties associated with the WOG-supplied analyses values. These uncertainties are to include uncertainties in computer program results and uncertainties resulting from plant-specific features not representative of the generic data group.

- If a licensee determines that the WOG alternative criteria are marginal for preventing unneeded RCP trip, it is recommended that a more discriminating plant-specific procedure be developed. Licensees should take credit for all equipment (instrumentation) available to the operators for which the licensee has sufficient confidence that it will be operable during the expected conditions.

The staff anticipates this information will be provided at a time consistent with completion of RCP trip criterion selection. The information should identify any differences between the WOG-provided generic analyses and the South Texas plants which have an influence on the results. The results are expected to be similar to information provided to the staff for other plants utilizing information provided by the WOG, and there is expected to be no significant impact upon staff conclusions.

Section B: Potential Reactor Coolant Pump Problems

Subsection B1

- Assure that containment isolation, including inadvertent isolation, will not cause problems if it occurs for non-LOCA transients and accidents.

  (a) Demonstrate that, if water services needed for RCP operations are terminated, they can be restored fast enough once a non-LOCA situation is confirmed to prevent seal damage or failure.

  (b) Confirm that containment isolation with continued pump operation will not lead to seal or pump damage or failure.

Essential water services for RCP operation are stated to continue under accident conditions involving containment isolation (CI). Seal injection is only isolated when a CI signal is present concurrent with low charging header pressure. Component cooling water (CCW) to the RCP thermal barrier heat exchangers is continued independent of CI, and is stated to be lost only on a low CCW surge tank level. The applicant further states that the only credible condition which would interrupt both of the sources of cooling water to the RCPs is a loss of offsite power, where the RCP motors are deenergized. Cooling and injection water are stated to be restored within seconds following start of the diesel generators.

The applicant's approach to seal support is to establish that seal injection is present unless there are major, multiple equipment failures. Typically, such a situation would result in a low header pressure, which in turn would be followed by valve closure of the supply lines. The case with cooling water to the RCP components is similar. This is normally unaffected by all levels of containment isolation.

Response to a loss of seal injection is to continue to operate the RCPs while attempting to restore seal injection flow in a controlled manner. RCP trip would be accomplished if a bearing or #1 seal inlet temperature exceeds established setpoints. Response to loss of cooling water to the oil coolers is to follow the RCP bearing temperatures while continuing to operate the RCPs. RCP trip would be initiated if temperature operational limits were exceeded. Loss of cooling water to the thermal barrier heat exchangers would not be a reason for RCP trip as long as a sufficient flow rate existed from the seal injection region into the RCS, as determined from the injection rate and leakoff rate instrumentation.

RCP restart following trip provides a full consideration to the effects of initiation of cooling water to a hot RCP component, and the potential thermal stresses which could be induced. The issues covered in Subsection B1 are adequately addressed.

Subsection B2

- Identify the components required to trip the RCPs, including relays, power supplies, and breakers. Assure that RCP trip, when necessary, will occur. Exclude extended RCP operation in a voided system where pump head is more than 10% degraded unless analyses or tests can justify pump and pump-seal integrity when operating in voided systems. If necessary, as a result of the location of any critical component, include the effects of adverse containment conditions on RCP trip reliability. Describe the basis for the adverse containment parameters selected.

All components associated with RCP trip are stated to be located outside containment, and therefore are not affected by adverse containment conditions. The breakers are located in the turbine building. The applicant has further stated that there are no relays outside of the breaker enclosures. Everything is within these enclosures with the exception of the control room switches and interconnecting wiring. The review of pipe breaks mentioned previously will encompass consideration of this area.

If an operator were to attempt an RCP trip from the control room, and it was unsuccessful, then trip would be accomplished locally at the breakers. The required time would be approximately 5 minutes. There are no locked doors to impede travel from the control room to the breakers, and an adverse environment that would interfere with trip operation is not anticipated.

Operation in a voided system is guided by the WOG Revision 1 Emergency Operating Guidelines. Note that under this guidance, RCPs are to be restarted under severe ICC conditions.

These operations are a part of operator training.

The staff concludes that the applicant has adequately addressed the issues identified in this subsection with the exception of the above identified pipe break review.

Section C: Operator Training and Procedures (RCP Trip)

Subsection C1

- Describe the operator training program for RCP trip. Include the general philosophy regarding the need to trip pumps versus the desire to keep pumps running. Also cover priorities for actions after engineered safety features actuation.

- Assure that training and procedures provide direction for use of individual steam generators with and without operating RCPs.

- Assume manual RCP trip does not occur earlier than 2 minutes after the RCP-trip setpoint is reached.

- Determine the time available to the operator to trip the RCPs for the limiting cases if manual RCP trip is proposed. Best estimate calculational procedures should be used. The applicant should identify and justify most-probable plant conditions, although the staff will accept conservative estimates in the absence of justifiable most-probable conditions.

- Justify that the time available to trip the RCPs is acceptable if it is less than the Draft ANSI Standard N660. If this is the case, then address the consequences if RCP trip is delayed. Also develop contingency procedures and make them available for the operator to use in case the RCPs are not tripped in the preferred time frame.

The applicant, by letter dated May 30, 1986, has clearly identified an understanding of the background and need for RCP trip under some conditions and the need for RCP operation under other conditions.

The staff has not reached final conclusion in regard to whether RCPs should be tripped or left running in the event they should have been tripped, and were not. As previously identified, the staff will follow up on this item on a generic basis at a later time. No further applicant action is needed at this time.

Subsection C2

- Identify those procedures which include RCP trip related operation:

  (a) RCP trip using WOG alternate criteria
  (b) RCP restart
  (c) decay heat removal by natural circulation
  (d) primary system void removal
  (e) use of steam generators with and without RCPs operating trip
  (f) RCP for other reasons

- Ensure that emergency operating procedures exist for the timely restart of the RCPs when conditions warrant.

The applicant has presented a summary listing of selected procedures which are stated to be based upon the WOG Guidelines. This list is sufficient to establish that adequate procedural information is available.

Conclusions

The applicant has satisfactorily addressed each of the points identified in GL 85-12. Further, the applicant has considered items pertinent to RCP trip and operation which are in addition to the requirements of GL 85-12. The applicant has not completed the work to select the RCP trip criterion, but has progressed sufficiently for the staff to conclude that either RCS pressure or RCS pressure to SG secondary side pressure differential will be acceptable. Although several other items to support the resolution of the RCP trip issue are also incomplete, the staff expects that the final results will not change its acceptance of the applicant's submittals pertinent to RCP trip. The following confirmatory information is required from the applicant:

(1) selected RCP trip criteria, including numerical values, final uncertainties, final calculated results, and a comparison of the selected criteria to the calculations which illustrates application of the uncertainties and establishes the separation between LOCA and non-LOCA events (Section A and Subsections A1, A2, and A3)

(2) impact of an adverse containment atmosphere on RCS pressure indication as determined by the ongoing program (Subsections A1 and A2)

(3) identification of the specific instrumentation used to determine the need for RCP trip (Subsection A1)

(4) summary of the program to evaluate pipe whip and fluid jets insofar as impact upon RCP trip is concerned (Subsection A2)

(5) definition of the radiation environment associated with an adverse containment (Subsection A2)

(6) results of the program to evaluate pipe breaks insofar as it impacts upon RCP trip considerations (Subsection B2)

15.8 Anticipated Transients Without Scram

15.8.1 ATWS Rule--ATWS Mitigation Systems

In the SER, the staff stated it was reviewing PWR generic ATWS designs. The staff has since reviewed the Westinghouse Topical Report WCAP-10858, "AMSAC Generic Design Package," and has concluded that the generic designs presented in WCAP-10858 adequately meet the requirements of 10 CFR 50.62 and follow the staff's review guidelines. The staff has not completed its review of the South Texas design for compliance with the ATWS rule; however, staff review and approval are not required for plant licensing. The staff will review the South Texas plant-specific design in accordance with the plant-specific review schedule. The staff's evaluation of the South Texas ATWS design will be provided in a future supplement to this SER. This is an update of Confirmatory Item 29.

15.8.2 Generic Letter 83-28--Actions

On February 25, 1983, both of the scram circuit breakers at Unit 1 of the Salem Nuclear Power Plant failed to open on receipt of an automatic reactor trip signal from the reactor protection system. This incident occurred during plant startup, and the reactor was tripped manually by the operator about 30 seconds after the initiation of the automatic trip signal. The staff determined that the failure of the circuit breakers was related to the sticking of the undervoltage trip attachment. Before this incident, on February 22, 1983, an automatic trip signal was generated on the basis of steam generator low-low level during plant startup at Unit 1 of the Salem Nuclear Power Plant. In this case, the reactor was tripped manually by the operator almost simultaneously with the automatic trip. Following these incidents, on February 28, 1983, the NRC Executive Director for Operations directed the NRC staff to investigate and report on the generic implications of these occurrences at Unit 1 of the Salem Nuclear Power Plant. The results of the staff's inquiry into the generic implications of the incidents at the Salem unit are reported in NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant." As a result of this investigation, the Commission requested (by Generic Letter 83-28 dated July 8, 1983) all licensees of operating reactors, applicants for an operating license, and holders of construction permits to respond to certain generic concerns. These concerns are categorized into four areas of action: (1) post-trip review, (2) equipment classification and vendor interface, (3) post-maintenance testing, and (4) reactor trip system reliability improvements. On March 7, 1986, the staff transmitted to the applicant a summary of the review status on the information submitted in response to GL 83-28.

(1) Post-Trip Review

Action Item 1.1: Program Description and Procedures

The following review guidelines were developed after initial evaluation of several utility responses to Item 1.1 of Generic Letter 83-28 and incorporate the best features of these submittals. These review guidelines represent a "good practices" approach to post-trip review. The staff has reviewed the applicant's response to Item 1.1 against these guidelines:

A. The licensee or applicant should have systematic safety assessment procedures established that will ensure that the following restart criteria are met before restart is authorized.

   - The post-trip review team has determined the root cause and sequence of events resulting in the plant trip.
   - Near-term corrective actions have been taken to remedy the cause of the trip.
   - The post-trip review team has performed an analysis and determined that the major safety systems responded to the event within specified limits of the primary system parameters.
   - The post-trip review has not resulted in the discovery of a potential safety concern (e.g., the root cause of the event occurs with a frequency significantly larger than expected).
   - If any of the above restart criteria are not met, then an independent assessment of the event is performed by the Plant Operations Review Committee (PORC), or another designated group with similar authority and experience.

B. The responsibilities and authorities of the personnel who will perform the review and analysis should be well defined.

   - The post-trip review team leader should be a member of plant management at the shift supervisor level or above and should hold or should have held a Senior Reactor Operator (SRO) license on the plant. The team leader should be charged with overall responsibility for directing the post-trip review, including data gathering and data assessment and should have the necessary authority to obtain all personnel and data needed for the post-trip review.
   - A second person on the review team should be a Shift Technical Advisor (STA) or should hold a relevant engineering degree with special transient analysis training.
   - The team leader and the STA (Engineer) should concur on a decision/recommendation to restart the plant. A nonconcurrence from either of these persons should be enough to prevent restart until the trip has been reviewed by the PORC or equivalent organization.

C. The licensee or applicant should indicate that the plant response to the trip event will be evaluated and a determination made as to whether the plant response was within acceptable limits. The evaluation should include:

- A verification of the proper operation of plant systems and equipment by comparison of the pertinent data obtained during the post-trip review to the applicable data provided in the FSAR.

- An analysis of the sequence of events to verify the proper functioning of safety related and other important equipment. Where possible, comparisons with previous similar events should be made.

D. The licensee or applicant should have procedures in place to ensure that all physical evidence necessary for an independent assessment is preserved.

E. Each licensee or applicant should provide in its submittal, copies of the plant procedures which contain the information required in Items A through D. As a minimum, these should include the following:

- The criteria for determining the acceptability of restart.

- The qualifications, responsibilities, and authorities of key personnel involved in the post-trip review process.

- The methods and criteria for determining whether the plant variables and system responses were within the limits as described in the FSAR.

- The criteria for determining the need for an independent review.

By letters dated February 26 and June 28, 1985, the applicant provided information regarding its Post-Trip Review Program. The staff has evaluated the applicant's program against the guidelines described above for Action Item 1.1 of GL 83-28.

A. With regard to the criteria for determining the acceptability of restart, the applicant will cause each unscheduled reactor trip to be reviewed and classified into one of three categories:

Category 1 - The cause of the trip has been identified. Safety-related equipment functioned properly.

Category 2 - The cause of the trip has been identified. Some safety-related equipment may not have functioned properly. The malfunctioning safety-related equipment must be corrected to comply with Technical Specification requirements before startup.

Category 3 - The cause of the reactor trip is unknown or safety-related equipment did not function properly and the malfunction has not been corrected.

The Shift Technical Advisor is responsible for obtaining the necessary information to perform the review, performing the initial review, and providing the review results to the Shift Supervisor for review and approval. The Shift Supervisor initiates corrective actions as deemed necessary and may authorize startup following a Category 1 trip. For a Category 2 trip, the Shift Supervisor provides recommendations to the Plant Manager regarding restart. The Plant Manager's approval is required for startup following a Category 2 or Category 3 trip. The Plant Operations Review Committee (PORC) will review all trips, and will review Category 3 trips prior to restart. The PORC provides its recommendations to the Plant Manager regarding corrective actions needed and the readiness of the plant for restart. The staff finds that the applicant's criteria for determining the acceptability for restart are in conformance with the guidelines described above and, therefore, are acceptable.

B. As summarized in item A above, the responsibilities and authorities of the personnel performing the review are well defined. Both the STA and the Shift Supervisor must agree on restart following a Category 1 trip. The Plant Manager approves restart following a Category 2 or Category 3 trip. The PORC reviews all trips, but must review Category 3 trips before restart is authorized. The staff finds that the applicant's chain of command for responsibility for post-trip review and evaluation are in conformance with the guidelines above and are acceptable.

C. The applicant has described the information that will be collected and evaluated during the performance of the post-trip review. The information will be analyzed for protection system and overall plant response, and will be compared as necessary with expected plant response as described in the FSAR, Technical Specifications, and with the plant response to previous reactor trips. The staff finds that the applicant's plans for post-trip analysis and evaluation are in conformance with the guidelines above and are, therefore, acceptable.

D. As noted in item A above, the applicant has stated that if the cause of the reactor trip is unknown, or if safety-related equipment did not function properly and the malfunction has not been corrected, the trip must be reviewed by the PORC before restart is authorized. The applicant has stated that trip review reports and supporting documentation will be retained as quality assurance records. The staff finds that the applicant's criteria for independent assessment of unscheduled reactor trips and for preservation of documentation relating to unscheduled trips are in conformance with the guidelines above and are, therefore, acceptable.

E. The applicant has stated that instructions on performing post-trip reviews, including what information to obtain and how to obtain it, instructions on how to categorize unscheduled trips, who has authority to authorize restart, forms to summarize the required information, checklists to guide the reviewers, and instructions to transmit trip reviews to the PORC will be incorporated in a plant procedure which will be approved before fuel load. The staff will confirm the existence and acceptability of this procedure during an audit visit to the plant site and will report the results of its confirmation review in a future SER supplement.

On the basis of its review, and subject to later review and confirmation of the procedure governing post-trip review, the staff finds the applicant's program for conducting post-trip reviews acceptable.

Action Item 1.2: Data and Information Capability

The following review guidelines were developed after initial evaluation of the various utility responses to Item 1.2 of GL 83-28 and incorporate the best features of these submittals. These review guidelines represent a "good practices" approach to post-trip review. The staff has reviewed the applicant's response to Item 1.2 against these guidelines:

A. The equipment that provides the digital sequence-of-events (SOE) record and the analog time-history records of an unscheduled shutdown should provide a reliable source of the necessary information to be used in the post-trip review. Each plant variable which is necessary to determine the cause and progression of the events following a plant trip should be monitored by at least one recorder (such as an SOE recorder or a plant process computer) for digital parameters; and strip charts, a plant process computer, or analog recorder for analog (time-history) variables. Performance characteristics guidelines for SOE and time history recorders are as follows:

   - Each SOE recorder should be capable of detecting and recording the sequence of events with a sufficient time discrimination capability to ensure that the time responses associated with each monitored safety-related system can be ascertained, and that a determination can be made as to whether the time response is within acceptable limits based on FSAR Chapter 15 accident analyses. The recommended guidelines for the SOE time discrimination is approximately 100 milliseconds. If current SOE recorders do not have this time discrimination capability, the applicant should show that the current time discrimination capability is sufficient for an adequate reconstruction of the course of the reactor trip and post-trip events. At a minimum this should include the ability to adequately reconstruct the transient and accident scenarios presented in Chapter 15 of the plant FSAR.

   - Each analog time history data recorder should have a sample interval small enough so that the incident can be accurately reconstructed following a reactor trip. At a minimum, the applicant should be able to reconstruct the course of the transient and accident sequences evaluated in the accident analysis of Chapter 15 of the plant FSAR. The recommended guideline for the sample interval is 10 seconds. If the time-history equipment does not meet this guideline, the applicant should show that the time-history capability is sufficient to accurately reconstruct the transient and accident sequences presented in Chapter 15 of the FSAR. To support the post-trip analysis of the cause of the trip and the proper functioning of involved safety-related equipment, each analog time history data recorder should be capable of updating and retaining information from approximately 5 minutes before the trip until at least 10 minutes after the trip.

   - All equipment used to record SOE and time-history information should be powered from a reliable and non-interruptible power source. The power source used need not be a Class 1E source.

B. The SOE and time-history recording equipment should monitor sufficient digital and analog parameters, respectively, to ensure that the course of the reactor trip and post-trip events can be reconstructed. The parameters monitored should provide sufficient information to determine the root cause of the unscheduled shutdown, the progression of the reactor trip, and the response of the plant parameters and protection and safety systems to the unscheduled shutdowns. Specifically, all input parameters associated with reactor trips, safety injections and other safety-related systems as well as output parameters sufficient to record the proper functioning of these systems should be recorded for use in the post-trip review. The parameters deemed necessary, at a minimum, to perform a post-trip review that would determine if the plant remained within its safety limit design envelope are presented in Table 15.7. They were selected on the basis of staff engineering judgment following a complete evaluation of utility submittals. If the applicant's SOE recorders and time-history recorders do not monitor all of the parameters suggested in these tables the applicant should show that the existing set of monitored parameters are sufficient to establish that the plant remained within the design envelope for the accident conditions analyzed in Chapter 15 of the plant FSAR.

C. The information gathered by the SOE and time-history recorders should be stored in a manner that will allow for data retrieval and analysis. The data may be retained in either hard copy (e.g., computer printout, strip chart record) or in an accessible memory (e.g., magnetic disc or tape). This information should be presented in a readable and meaningful format, taking into consideration good human factors practices such as those outlined in NUREG-0700.

D. Retention of data from all unscheduled shutdowns provides a valuable reference source for the determination of the acceptability of the plant vital parameter and equipment response to subsequent unscheduled shutdowns. Information gathered during the post-trip review is to be retained for the life of the plant for post-trip review comparisons of subsequent events.

By letters dated November 3, 1983, February 26, 1985, and May 30, 1986, the applicant provided information regarding its Post-Trip Review Program data and information capabilities for South Texas Project. The staff has evaluated the applicant's submittals against the review guidelines described above for Action Item 1.2 of Generic Letter 83-28. A brief description of the applicant's responses and the staff's evaluation of the response against each of the review guidelines follows:

A. The applicant has described the performance characteristics of the equipment used to record the SOE and time-history data needed for post-trip review. On the basis of its review of the applicant's submittals, the staff finds that the SOE recorder and time-history recorder characteristics conform to review guideline A above, and are acceptable.

B. The applicant has established and identified the parameters to be monitored and recorded for post-trip review. On the basis of its review, the staff finds that the parameters selected by the applicant will include all but one of those identified in Table 15.7. Although control rod position is not included as a parameter, required sequence information on rod position is derived from the reactor trip breaker position inputs. Additional rod position information can be obtained from the long-term data storage files provided by the Proteus and ERFDADS computer systems. The staff finds this acceptable. Consequently, the staff finds that the applicant's selection of parameters meets the intent of review guideline B above and is, therefore, acceptable.

C. The applicant described the means for storage and retrieval of the information gathered by the SOE and time-history recorders, and for the presentation of this information for post-trip review and analysis. On the basis of its review, the staff finds that this information will be presented in a readable and meaningful format, and that the storage, retrieval and presentation conform to review guideline C above.

D. The applicant's submittal of May 30, 1986, indicates that the data and information used during post-trip reviews will be retained in an accessible manner for the life of the plant. On the basis of this information, the staff finds that the applicant's program for data retention conforms to review guideline D above, and is acceptable.

On the basis of its our review of the applicant's submittals, the staff concludes that the applicant's post-trip review data and information capabilities for South Texas Project are acceptable.

(2) Equipment Classification and Vendor Interface

Action Item 2.1: Reactor Trip System Components

Action Item 2.1 (Part 1): Equipment Classification

Item 2.1 (Part 1) requires the applicant to confirm that all reactor trip system components are identified, classified, and treated as safety related as indicated in the following statement:

> Licensees and applicants shall confirm that all components whose functioning is required to trip the reactor are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders, and parts replacement.

The applicant responded to the requirements of Item 2.1 (Part 1) with a submittal dated June 28, 1985. The applicant stated in this submittal that all components that are required to perform the reactor trip function were reviewed to verify that these components are classified as safety-related equipment. The classification is designated in design documents and the plant Q-list. Maintenance, work orders, and parts replacement require identification of safety classification before approval.

On the basis of its review of these responses, the staff finds the applicant's statements confirm that a program exists for identifying, classifying, and treating components that are required for performance of the reactor trip function as safety related. This program meets the requirements of Item 2.1 (Part 1) of GL 83-28, and is, therefore, acceptable. The applicant also has a computerized maintenance system under development. This system, which deals with all safety-related components, will be reviewed by the staff during the forthcoming review of Item 2.2.1. EG&G Idaho, Inc., NRC's consultant on Item 2.1 (Part 1), provided the staff with a technical evaluation report which is reproduced in this supplement in Appendix N.

Action Item 2.1 (Part 2): Vendor Interface Program

Item 2.1 (Part 2) requires that, for all reactor trip system components, licensees and applicants establish, implement, and maintain a continuing program to ensure that vendor information is complete, current, and controlled throughout the life of the plant, and appropriately referenced or incorporated in plant instructions and procedures. The applicant responded to the requirements of Item 2.1 (Part 2) in submittals dated February 26 and June 28, 1985. Part 2 of this action item is under staff review. The results of this review will be reported in a future supplement to the SER.

Action Item 2.2: Programs for All Safety-Related Components

Action Item 2.2.1: Equipment Classification

Item 2.2.1 requires that for equipment classification, licensees and applicants describe in considerable detail their program for ensuring that all components of safety-related systems necessary for accomplishing required safety functions are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders, and replacement parts.

Action Item 2.2.2: Vendor Interface Program

Item 2.2.2 requires that licensees and applicants establish, implement, and maintain a continuing vendor interface program which ensures that vendor information for safety-related components is complete, current, and controlled throughout the life of their plants, and is appropriately referenced or incorporated in plant instructions and procedures.

The applicant responded to the requirements of the two parts of Item 2.2 in submittals dated February 26, 1985 and June 28, 1985. This action item is under staff review and the results will be reported in a future supplement to the SER.

(3) Post-Maintenance Testing

Action Items 3.1.1, 3.1.2, 3.1.3: Reactor Trip System Components

Item 3.1.1 requires that licensees and applicants submit the results of their review of test and maintenance procedures and Technical Specifications to assure that post-maintenance operability testing of safety-related components in the reactor trip system is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

Item 3.1.2 requires that licensees and applicants submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications, where required.

The applicant responded to the requirements of Items 3.1.1 and 3.1.2 in a submittal dated June 28, 1985. These two action items are under staff review.

The requirements for Items 3.1.3 and 3.2.3 are identical, with the exception that Item 3.1.3 applies these requirements to the reactor trip system components and Item 3.2.3 applies them to all other safety-related components. Because of this similarity, the applicant responses to both items were evaluated together.

Items 3.1.3 and 3.2.3 require that licensees and applicants identify, if applicable, any post-maintenance test requirements in existing Technical Specifications which can be demonstrated to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval.

The applicant for the South Texas Project, Units 1 and 2, responded to these requirements with submittals dated June 28, 1985, and January 28, 1986. The applicant stated in these submittals that there were no post-maintenance testing requirements in Technical Specifications for either the reactor trip system or other safety-related components which degraded safety.

On the basis of the applicant's statement that no post-maintenance test requirements were found in Technical Specifications that degraded safety, the staff finds the applicant's responses for Items 3.1.3 and 3.2.3 acceptable. NRC's consultant, EG&G, Inc. for Items 3.1.3 and 3.2.3, provided the staff with a Technical Evaluation Report. This report is provided as Appendix O to this supplement.

Action Items 3.2.1, 3.2.2 and 3.2.3: All Other Safety-Related Components

Item 3.2.1 requires that licensees and applicants submit a report documenting the extending of test and maintenance procedures and Technical Specifications review to assure that post-maintenance operability testing of all safety-related equipment is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

Item 3.2.2 requires that licensees and applicants submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications where required.

The applicant responded to the requirements of Items 3.2.1 and 3.2.2 in a submittal dated June 28, 1986. These action items are under staff review.

Item 3.2.3 was evaluated with Item 3.1.3 and was found acceptable.

(4) Reactor Trip System Reliability Improvements

Action Item 4.1: Vendor-Related Modifications

and

Action Item 4.2: Preventative Maintenance and Surveillance Program for Reactor Trip Breakers

Item 4.1 requires licensees and applicants to verify that all vendor-recommended reactor trip breaker modifications have been implemented. Item 4.2 required them to submit a description of their preventive maintenance and surveillance program to ensure reliable reactor trip breaker operation. The description of the submitted program was to include the following:

Item 4.1 requires that all vendor-recommended reactor trip breaker modifications shall be reviewed to verify that either each modification has in fact been implemented, or that a written evaluation of the technical reasons for not implementing a modification exists.

Item 4.2.1 requires a planned program of periodic maintenance, including lubrication, housekeeping, and other items recommended by the equipment supplier.

Item 4.2.2 requires trending of parameters affecting operation and measured during testing to forecast degradation of operation.

The applicant submitted responses to Item 4.1 on February 26 and June 28, 1985, and to Items 4.2.1 and 4.2.2 on June 28, 1985. The staff review presents an evaluation of the adequacy of the applicant's responses and of preventive maintenance and surveillance programs for reactor trip breakers (RTBs).

The primary source for periodic maintenance program criteria is Westinghouse Maintenance Program Manual for DS-416 Reactor Trip Circuit Breakers, Rev. 0. This document was prepared for the Westinghouse Owners Group and is the breaker manufacturer's recommended maintenance program for the DS-416 breaker. It provides specific direction with regard to schedule, inspection and testing, cleaning, lubrication, corrective maintenance, and recordkeeping. The document was reviewed to identify those items that contribute to breaker trip reliability consistent with Generic Letter 83-28. Those items identified for maintenance at 6-month intervals (or when 500 breaker operations have been counted, whichever comes first) that should be included in the licensee's RTB maintenance program are:

(1) general inspection to include checking of breaker's cleanliness, all bolts and nuts, pole bases, arc chutes, insulating link, wiring, and auxiliary switches

(2) retaining rings inspection, including those on the undervoltage trip attachment (UVTA)

(3) arcing and main contacts inspection as specified by the Westinghouse Maintenance Manual

(4) UVTA check as specified by the Westinghouse Maintenance Manual, including replacement of UVTA if dropout voltage is greater than 60% or less than 30 of rated UVTA coil voltage

(5) shunt trip attachment (STA) check as specified by the Westinghouse Maintenance Manual

(6) lubrication as specified by the Westinghouse Maintenance Manual

(7) functional check of the breaker's operation before returning it to service.

The licensee's RTB periodic maintenance should also include, on a refueling interval basis:

(1) pre-cleaning insulation resistance measurement and recording

(2) RTB dusting and cleaning

(3) post-cleaning insulation resistance measurement and recording, as specified by the Westinghouse Maintenance Manual

(4) inspection of main and secondary disconnecting contacts, bolt tightness, secondary wiring, mechanical parts, cell switches, instruments, relays, and other panel-mounted devices

(5) UVTA trip force and breaker load check as specified by the Westinghouse Maintenance Manual

(6) measurement and recording of RTB response time for the undervoltage trip

(7) Functional test of the breaker before returning it to service, as specified by the Westinghouse Maintenance Manual.

Item 4.2.2 specifies that the applicant's preventive maintenance and surveillance program is to include trending of parameters affecting operation and measured during testing to forecast degradation of operation. The parameters measured during the maintenance program described above which are applicable for trending are undervoltage trip attachment dropout voltage, trip force, response time for undervoltage trip and breaker insulation resistance. The staff position is that the above parameters are acceptable and recommended trending parameters to forecast breaker operation degradation or failure. If subsequent experience indicates that any of these parameters is not useful as a tool to anticipate failures or degradation, the licensee may, with justification and NRC approval, elect to remove that parameter from those to be tracked.

The applicant stated in a submittal dated June 28, 1985, that Westinghouse will replace the undervoltage attachment on the DS-416 reactor trip breakers in the South Texas Project, Units 1 and 2. Work for each of the two units will be completed before their prospective fuel loading.

The staff finds that the applicant has committed to implement all vendor-related modifications before the fuel loading. Therefore, the applicant's position on Item 4.1 is acceptable.

The applicant has confirmed by submittal dated June 28, 1985, that its periodic maintenance program includes all of the seven items identified for maintenance at 6-month intervals and the other seven identified at the refueling intervals. The 14 items are mentioned under evaluation criteria listed above. Therefore, the staff finds the applicant position on Item 4.2.1 acceptable.

The applicant is committed to develop trending procedures to include the undervoltage device drop-out voltage, trip lever force, breaker response time for undervoltage trip and the breaker insulation resistance. The applicant states that Nuclear Plant Operations Department (NPOD) will collect and trend the data. NPOD will specify changes to the periodic maintenance program based upon trend analysis results.

The staff finds the applicant's commitment adequate. The applicant's position on Item 4.2.2 is acceptable.

On the basis of a review of the applicant's responses, the staff finds the applicant's position on Items 4.1, 4.2.1 and 4.2.2 acceptable.

Item 4.2.3 requires life testing of the breakers (including the trip attachments) on an acceptable sample size.

Item 4.2.4 requires periodic replacement of breakers or components consistent with demonstrated life cycles.

The applicant responded to the requirements of Items 4.2.3 and 4.2.4 in submittals dated June 28, 1985, and May 30, 1986, and is awaiting staff review and approval of Westinghouse Owners Group (WOG) report, WCAP-10835, "Report of the DS-416 Reactor Trip Breaker Undervoltage and Shunt Trip Attachments Life Cycle Tests." This WOG submittal is under staff review.

Action Item 4.3: Automatic Actuation of Shunt Trip Attachment for Westinghouse and B&W Plants

Item 4.3 requires that modifications be made to improve the reliability of the reactor trip system by implementation of an automatic actuation of the shunt trip attachment on the reactor trip breakers. By letter dated October 14, 1985, the applicant provided responses to the plant-specific questions identified by the staff in its August 10, 1983, safety evaluation report of the generic Westinghouse design. The staff has reviewed the applicant's proposed design for the automatic actuation of the reactor trip breaker shunt trip attachments and finds it acceptable.

The applicant has not specified the implementation date for these modifications.

The following required plant-specific information items were identified based on the staff's review of the WOG proposed generic design for this modification:

(1) Provide the electrical schematic/elementary diagrams for the reactor trip and bypass breakers showing the undervoltage and shunt coil actuation circuits as well as the breaker control (e.g., closing) circuits, and circuits providing breaker status information/alarms to the control room.

The applicant provided the electrical schematic diagrams for the reactor trip and bypass breakers showing the undervoltage and the shunt trip circuits. The design of the electrical circuits have been reviewed and found to be consistent with the WOG generic proposed design which was previously reviewed and approved by the staff. The staff finds this acceptable.

(2) Identify the power sources for the shunt trip coils. Verify that they are Class 1E and that all components providing power to the shunt trip circuitry are Class 1E and that any faults within non-Class 1E circuitry will not degrade the shunt trip function. Describe the annunciation/indication provided in the control room upon loss of power to the shunt trip circuits. Also describe the overvoltage protection and/or alarms provided to prevent or alert the operator(s) to an overvoltage condition that could affect both the undervoltage (UV) coil and the parallel shunt trip actuation relay.

Redundant Class 1E power sources are used for the shunt trip actuation of the reactor trip breakers and for the shunt trip of the bypass breakers. Class 1E circuitry is separated from non-Class 1E circuitry. Therefore, credible faults within non-Class 1E circuitry will not degrade the shunt trip function. This is in accordance with Regulatory Guide 1.75 and is, therefore, acceptable.

The breaker position status lights are used to supervise the availability of power to the shunt trip circuits. The red light which is connected in series with the shunt coil and the "a" auxiliary contact indicates that the breaker is closed and also indicates that the power is available to the shunt trip device and, therefore provides detectability of power failure to the shunt trip coil.

Normally the shunt trip coils in the reactor trip breakers are in de-energized condition. When the trip breakers are closed, the red lamp current (approximately 50 ma) flows through the trip coil to monitor the circuit continuity but is not large enough to actuate the trip coil armature. Since the current through the shunt trip coils is interrupted when the breaker trips, energization of the shunt trip coil is only momentary. The maximum available voltage occurs during a battery equalizing charge at a maximum voltage of 115% of the nominal voltage. Due to the short duty cycle of the shunt trip coil, it can operate at this overvoltage condition without harmful effects.

The added shunt trip circuitry is powered from the reactor protection logic voltage supply (48 V dc). Components in the added shunt trip circuitry have been selected based on their ability to perform their intended function up to 115% of nominal voltage. The reactor protection logic voltage is provided with overvoltage protection set at 115% of nominal voltage.

On the basis of its review, the staff concludes that appropriate consideration has been given to the aspects of the design described above and the design is, therefore, acceptable.

(3) Verify that the relays added for the automatic shunt trip function are within the capacity of their associated power supplies and that the relay contacts are adequately sized to accomplish the shunt trip function. If the added relays are other than the Potter & Brumfield MDR series relays (P/N 2383A38 or P/N 955655) recommended by Westinghouse, provide a description of the relays and their design specifications.

The added relays for the automatic trip function are Potter and Brumfield MDR series relays P/N 955655. Westinghouse has verified that the relay contacts are adequately sized for the shunt trip function and are within the capacity of their associated power supplies. The staff finds this acceptable.

(4) State whether the test procedure/sequence used to independently verify operability of the undervoltage and shunt trip devices in response to an automatic reactor trip signal is identical to the test procedure proposed by the Westinghouse Owners Group (WOG). Identify any differences between the WOG test procedure and the test procedure to be used and provide the rationale/justification for these differences.

The applicant stated that the test procedures used to independently verify operability of the UV and shunt trip devices will be written and in place before fuel load. The procedures submitted by the WOG will be referenced during the development of its procedures. No major deviations from the WOG procedures are anticipated. The staff finds this commitment acceptable.

(5) Verify that the circuitry used to implement the automatic shunt trip function is Class 1E (safety related), and that the procurement, installation, operation, testing, and maintenance of this circuitry will be in accordance with the quality assurance criteria set forth in Appendix B to 10 CFR 50.

The applicant confirmed that the circuitry used to implement the automatic shunt trip function is Class 1E (safety related) and the procurement, installation, operation, testing and maintenance of this circuitry will be in accordance with the Westinghouse and South Texas Project quality assurance procedures which satisfy the quality assurance requirements of Appendix B to 10 CFR 50. The staff finds this acceptable.

(6) Verify that the shunt trip attachments and associated circuitry are/will be seismically qualified (i.e., be demonstrated to be operable during and after a seismic event) in accordance with the provisions of Regulatory Guide 1.100, Revision 1 (August 1977) which endorses IEEE Standard 344-1975, and that all non-safety-related circuitry/components in physical proximity to or associated with the automatic shunt trip function will not degrade this function during or after a seismic event.

The applicant stated that all components of the shunt trip and associated circuitry are incorporated within the reactor trip switchgear cabinets and are seismically qualified. The staff finds this acceptable.

(7) Verify that the components used to accomplish the automatic shunt trip function are designed for the environment where they are located.

The applicant noted that the components used to accomplish the automatic shunt trip function are designed for the environment where they are located. The staff finds this acceptable.

(8) Describe the physical separation provided between the circuits used to manually initiate the shunt trip attachments of the redundant reactor trip breakers. If physical separation is not maintained between these circuits, demonstrate that faults within these circuits cannot degrade both redundant trains.

The applicant confirmed that physical separation is maintained between redundant trains in the main control board, reactor trip switchgear and reactor protection logic for the shunt trip circuitry. Dual section manual reactor trip switches, with metal barriers between redundant train decks, are provided on the main control board. Shunt trip attachments interposing relays and their associated terminal blocks are mounted in separate metal enclosures. The reactor protection logic outputs for energizing the shunt trip interposing relays are housed in existing separate metal enclosures. Physical separation for field cabling between the redundant trains is maintained. The staff finds this meets the requirement of Regulatory Guide 1.75 and is, therefore, acceptable.

(9) Verify that the operability of the control room manual reactor trip switch contacts and wiring will be adequately tested before startup after each refueling outage. Verify that the test procedure used will not involve installing jumpers, lifting leads, or pulling fuses and identify any deviations from the WOG procedure. Permanently installed test connections (i.e., to allow connection of a voltmeter) are acceptable.

The applicant stated that the test procedures will be written to verify the operability of control room manual reactor trip switch contacts and wiring before startup after each refueling outage. The test procedures will not involve installing jumpers, lifting leads, or pulling fuses. The staff finds this commitment acceptable.

(10) Verify that each bypass breaker will be tested to demonstrate its operability before placing it into service for reactor trip breaker testing.

The applicant stated that the Technical Specifications were revised to include testing of the bypass breaker before placing it into service for reactor trip breaker testing. The staff finds this acceptable.

(11) Verify that the test procedure used to determine reactor trip breaker operability will also demonstrate proper operation of the associated control room indication/annunciation.

The applicant noted that the revised test procedures used to determine reactor trip breaker operability will demonstrate proper operation of the associated control room indication/annunciation. The staff finds this acceptable.

(12) Verify that the response time of the automatic shunt trip feature will be tested periodically and shown to be less than or equal to that assumed in the FSAR analyses or that specified in the Technical Specifications.

The applicant stated that Westinghouse has prepared a report of the reactor trip breaker UVTA and shunt trip attachment (STA) life cycle test which concludes that periodic testing for STA can be limited to verifying that it can trip the breaker with 70 V dc (minimum design voltage). Therefore, periodic testing of the automatic shunt trip feature response time is not required. The staff finds this acceptable.

(13) Propose Technical Specification changes to require periodic testing of the undervoltage and shunt trip functions and the manual reactor trip switch contacts and wiring.

The applicant submitted the proposed Technical Specification changes to require periodic testing of the undervoltage and shunt trip functions and the manual reactor trip switch contacts and wiring. The staff finds this in accordance with GL 85-09 and, therefore, acceptable. However, the applicant added ACTION 11 to Table 3.3-1, but failed to reference it in Item No. 19. Therefore, the staff required that the applicant add ACTION 11 to Item No. 19 of Table 3.3-1. The applicant in a submittal dated May 30, 1986, informed the staff that ACTION 11 is referenced in Item 19 of Table 3.3-1.

On the basis of a review of the applicant's response to the plant specific questions identified in the staff's evaluation of the Owners Group generic design modifications, the staff finds the modifications acceptable.

Action Item 4.4: Improvements in Maintenance and Test Procedures for B&W Plants

Because South Texas Project, Units 1 and 2, are Westinghouse designs, this action item is not applicable.

Action Items 4.5, 4.5.1, 4.5.2, and 4.5.3: System Function Testing

Item 4.5 requires on-line functional testing of the reactor trip system, including independent testing of the diverse trip features, be performed on all plants.

Item 4.5.1 requires that the diverse trip features to be tested include the breaker undervoltage and shunt trip features on Westinghouse plants.

Item 4.5.2 requires plants not currently designed to permit periodic on-line testing shall justify not making modifications to permit such testing. Alternatives to on-line testing proposed by licensees will be considered where special circumstances exist and where the objective of high reliability can be met in another way.

Item 4.5.3 requires existing intervals for on-line functional testing required by Technical Specifications be reviewed to determine that the intervals are consistent with achieving high reactor trip system availability when accounting for considerations such as:

(1) uncertainties in component failure rates
(2) uncertainty in common mode failure rates
(3) reduced redundancy during testing
(4) operator errors during testing
(5) component "wearout" caused by the testing

The applicant responded to the requirements of Item 4.5 in a submittal dated June 28, 1985. This action item is under staff review and results will be reported in a future supplement to the SER.

Licensees currently not performing periodic on-line testing shall determine appropriate test intervals as described above. Changes to existing required intervals for on-line testing as well as the intervals to be determined by those licensees not currently performing on-line testing shall be justified by information on the sensitivity of RTS availability to parameters such as the test intervals, component failure rates, and common-mode failure rates.

Table 15.7 PWR parameter list

SOE recorder	Time-history recorder	Parameter/Signal
(1) x		Reactor trip
(1) x		Safety injection
x		Containment isolation
(1) x		Turbine trip
x		Control rod position
(1) x	x	Neutron flux, power
x	x	Containment pressure
(2)	(2)	Containment radiation
	x	Containment sump level
(1) x	x	Primary system pressure
(1) x	x	Primary system temperature
(1) x		Pressurizer level
(1) x		Reactor coolant pump status
(1) x	x	Primary system flow
(3)		Safety injection; flow, pump/valve status
x		MSIV position
x	x	Steam generator pressure
(1) x	x	Steam generator level
(1) x	x	Feedwater flow
(1) x	x	Steam flow
(3)		Auxiliary feedwater system; flow, pump/valve status
x		ac and dc system status (bus voltage)

Table 15.7 PWR parameter list

SOE recorder	Time-history recorder	Parameter/Signal
x		Diesel generator status (start/stop, on/off)
x		PORV position

(1) Trip parameters.

(2) Parameter may be monitored by either an SOE or time history recorder.

(3) Acceptable recorder options are; (a) system flow recorded on an SOE recorder, (b) system flow recorded on a time history recorder, or (c) equipment status recorded on an SOE recorder.

16 TECHNICAL SPECIFICATIONS

On the basis of additional information provided in this Supplemental Safety Evaluation Report (SSER) in Sections 2.4.1.4, 5.2.2.1, and 6.3.1, Table 16.1, "Listing of Technical Specification items," has been revised in this supplement. Additional information is added to item 1 and three additional items (20, 21 and 22) are added.

Table 16.1 Listing of Technical Specification items

Item		SER Section
(1)	LCO on essential cooling pond Water level in the essential cooling pond (ECP) falls below 25.5 ft MSL or the water temperature at the intake sides of the pond exceeds 95°F.	2.4.1.4
(2)	Surveillance of seismic instrumentation	3.7.4
(3)	LCO and surveillance requirements on leakage from pumps and valves	3.9.6
(4)	Monitoring of reactor coolant system flow at least every 24 hours	4.4.3.2
(5)	Verification of thermal design flow in startup testing	4.4.3.2
(6)	Final Technical Specifications on the inadequate core cooling system	4.4.6.3
(7)	Power-operated relief valve setpoint curve update, pumps' lockout for cold overpressure mitigation system, and primary-secondary temperature mismatch	5.2.2.2
(8)	Limiting conditions for operation on identified and unidentified leakage from reactor coolant pressure boundary	5.2.5
(9)	Periodic boron concentration measurements during cooldown	5.4.7.2
(10)	Operability requirements for reactor vessel head vent system	5.4.12
(11)	Surveillance requirements on sodium hydroxide in spray additive tanks	6.1.1
(12)	Operability testing of emergency core cooling system	6.3.6
(13)	Instrumentation and control setpoints	7.2.2.1
(14)	Response time testing of reactor trip breakers	7.2.2.2

Table 16.1 (Continued)

Item		SER Section
(15)	Compliance with Branch Technical Position PSB-1 (a) Voltage and time-delay setpoints (b) Test requirement to demonstrate operability of the automatic bypass and reinstatement features at least once every 18 months	8.3.1
(16)	Testing of diesel generators if the 2,000-hour rating is exceeded	8.3.1
(17)	Inservice inspection program for turbine valves	10.2
(18)	Independent review and audit by the NSRB	13.4.2.1
(19)	Valve positions to preclude boron dilution	15.4.6
(20)	NUREG-0737 Action Item II.K.3.3, "Reporting of SV and RV failure and challenges"	5.2.2.1
(21)	NUREG-0737 Action Item II.K.3.17, "Report on outages of emergency core cooling systems"	6.3.1
(22)	Required actions based on generic implications of Salem ATWS events (Generic Letter 83-28)	15.8.2

19 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

On May 29 and 30, 1986, a subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) met with representatives of the applicant and the NRC staff to consider the applicant's application for a license to operate the South Texas Project, Units 1 and 2. The meeting was held in Bay City, Texas. On June 5-7, 1986, at its 314th meeting, the full Advisory Committee on Reactor Safeguards met with representatives of the applicant and the staff to consider the application.

The ACRS recommended that the staff and applicant continue to resolve the open items. As a result of its review and meetings, the ACRS reported that, if due consideration is given to those items and to the following three items noted in its letter:

(1) environmental qualification of the residual heat removal pump for operation inside containment is case of an accident

(2) resolution of Construction Appraisal Team inspection findings

(3) testing and appropriate corrective measures to assure prevention of failures in the fuel oil piping and tubing by induced vibration resulting from extended operation of the diesel generators

there is reasonable assurance that the South Texas Project, Units 1 and 2, can be operated at power levels up to 3800 MWt without undue risk to the health and safety of the public. The ACRS letter from David A. Ward to Nunzio J. Palladino, dated June 10, 1986, is included as Appendix K to this supplement.

The staff has been reviewing and will continue to review the open items to ensure that they are resolved in a satisfactory manner before granting an operating license for South Texas Project, Units 1 and 2. This first supplement to the SER resolves 4 open items, provides an update to the evaluation on 2 open items, and adds 1 open item regarding thimble tube vibration (see Section 3.9.2.3). Further, Supplement 1 resolves 5 confirmatory items, provides an update to the evaluation on 3 confirmatory items, and adds 1 confirmatory item, a NUREG-0737 Action Item (II.K.3.5) which is being separated from Confirmatory Item 34 (see Section 15.6.5.1). Refer to Tables 1.4 and 1.5 for a complete list and for the status of open and confirmatory items.

The applicant responded to the other specific recommendations of the ACRS in a submittal dated June 27, 1986.

(1) Environmental Qualification of the Residual Heat Removal Pump

In an earlier submittal dated October 31, 1985, the applicant responded to an NRC staff question that the residual heat removal (RHR) system will be qualified for the environmental conditions resulting from a loss-of-coolant accident (LOCA) and main steamline break. This qualification would demonstrate that the RHR system would be capable of operating, as required, to provide for long-term reactor core cooling in the unlikely event of a small-break LOCA, an isolatable LOCA, or a main steamline/main feedline break inside reactor containment. In the SER, the staff carried a license condition (Table 1.6, License Condition 2) stating that postaccident qualification of the RHR system be completed before December 31, 1988, or the second refueling outage, whichever comes first. See SER Section 5.4.7.7 for further discussion.

Since issuance of the SER, the applicant by a submittal dated June 27, 1986, stated that existing environmental qualification documentation provides what the applicant believes to be sufficient basis for demonstrating the qualification of the RHR pump motors. The qualification envelope is based on consideration of the expected operating environment and anticipated operational requirement for the RHR system. The applicant is proceeding to document its conclusions and stated that documentation would be in place by September 1986. The NRC staff will verify this information in an audit and will provide an evaluation of RHR system environmental qualification in a future SER supplement.

(2) Construction Appraisal Team Inspection Findings

On the basis of the information provided by the Construction Appraisal Team (CAT) during its exit meeting which was held November 27, 1985, the applicant initiated actions to address the specific inspection findings and a number of improvements to enhance proper completion of the South Texas Project. These improvements were described in a submittal dated January 10, 1986.

The staff issued the CAT inspection report on February 5, 1986. The NRC CAT found that hardware and documentation for construction activities were generally in accordance with the requirements and license commitments. However, the NRC CAT did identify a number of hardware deficiencies that in most cases have resulted from construction program weaknesses. The applicant responded to these weaknesses identified in the transmittal letter and the Executive Summary (Appendix A to the CAT inspection report) and items listed as Potential Enforcement Actions (Appendix B to the CAT inspection report) by letter dated April 2, 1986. In another submittal (June 13, 1986), the applicant provided the staff with a description of a number of additional actions and programs which are having a beneficial effect on the successful completion of the South Texas Project.

All applicant submittals will be included as part of the staff's continuing review of this matter.

(3) Testing of Diesel Generator Fuel Oil Piping

The applicant, in previous responses to NRC staff questions, committed to a preoperational vibration testing program which included testing the diesel generator system. In addition, the applicant, in a submittal dated June 27, 1986, committed to augment this test program to obtain detailed data on the effects of diesel engine vibration on the fuel oil supply piping in order to respond to ACRS recommendations. By a submittal dated August 29, 1986, the applicant provided the test plan for vibration testing of the diesel fuel oil piping. Since the layout of the diesel generators and associated fuel oil piping is the same for both South Texas Project units, the applicant plans to use the results of the testing and any required modification on Unit 1 for

Unit 2 also. The applicant stated that tests will be completed in the fourth quarter of 1986 and any required modifications for Unit 1 will be completed in the first quarter of 1987 and for Unit 2 by Unit 2 fuel load. This test plan is under staff review. An evaluation will be reported in a future SER supplement.

APPENDIX A

CONTINUATION OF NRC STAFF RADIOLOGICAL REVIEW OF THE SOUTH TEXAS PROJECT

February 20, 1985*	Letter from applicant regarding interim response to NRC Generic Letter 83-28.
June 28, 1985*	Letter from applicant regarding response to NRC Generic Letter 83-28.
October 14, 1985*	Letter from applicant regarding response to NRC Generic Letter 83-28.
January 10, 1986*	Letter from applicant regarding management actions to address the implication of the recent NRC Construction Appraisal Team (CAT) inspection.
January 23, 1986*	Letter from applicant regarding response to TMI Action Plan Item II.K.3.5 and Generic Letter 85-12.
January 28, 1986*	Letter from applicant regarding response to Request for Additional Information Regarding NRC Generic Letter 83-28.
February 5, 1986*	Letter to applicant regarding CAT Inspections 50-498/85-21 and 50-499/85-19.
March 11, 1986*	Letter from applicant transmitting a revised auxiliary feedwater system reliability study.
March 19, 1986	Letter to applicant transmitting Draft Environmental Statement for South Texas Project (NUREG-1171).
March 20, 1986	Letter to applicant transmitting Generic Letter 86-07 concerning NUREG-1190 (loss of power and waterhammer event at San Onofre Unit 1).
March 23, 1986	Letter to applicant transmitting Generic Letter 86-08 concerning availability of Supplement 4 to NUREG-0933, "Prioritization of Generic Safety Issues."
March 26, 1986	Letter from applicant transmitting supplemental information on auxiliary feedwater requirements in the form of revised FSAR pages.

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*Although the dates of these letters precede this continuation of chronology, they are included here because they respond to issues discussed in this supplement.

March 26, 1986	Letter from applicant transmitting responses to NRC requests for additional information regarding Regulatory Guide 1.97.
March 31, 1986	Letter to applicant transmitting Generic Letter 86-09 concerning technical resolution of loop operation in BWRs and PWRs.
April 2, 1986	Letter from applicant regarding action taken in response to NRC CAT inspection.
April 8, 1986	Letter from applicant concerning South Texas Project Emergency Plan.
April 9, 1986	Letter from applicant transmitting FSAR Amendment 53. The Amendment consists of revised and updated information.
April 14, 1986	Letter from applicant concerning proposed change to FSAR Section 14.2.2.8 on startup personnel qualifications.
April 17, 1986	Letter from applicant transmitting responses to NUREG-0737 Items II.K.1.5, II.K.1.10, II.K.1.17.
April 17, 1986	Letter from applicant concerning status of review of South Texas Project information on Generic Letter 85-12.
April 21, 1986	Letter from applicant concerning South Texas Project - Site Specific Terrain Adjustment Factors.
April 28, 1986	Letter to Westinghouse withholding from public disclosure supplemental information regarding fuel assembly response to seismic/LOCA forces CAW-86-013 on the South Texas docket.
April 28, 1986	Letter from applicant concerning FES Impingement/Entrainment Monitoring Program.
April 29, 1986	Letter to applicant transmitting 20 copies of the South Texas Project Safety Evaluation Report - NUREG-0781.
May 2, 1986	Letter from applicant concerning revision to FSAR Section 3.6 - Two Phase Jet Criteria.
May 2, 1986	Letter from applicant concerning Security Personnel Training and Qualification Plan.
May 7, 1986	Letter from applicant concerning Emergency Plan tables and figures.
May 8, 1986	Letter from applicant transmitting Preliminary Scoping Study Results.
May 8, 1986	Letter from applicant responding to Policy Statement on Engineering Expertise on Shift (Generic Letter 86-04).
May 8, 1986	Letter from applicant concerning comments on the Draft Environmental Statement.

May 8, 1986	Letter to applicant concerning Exemption Request to GDC 4.
May 19, 1986	Letter to applicant concerning second audit of the South Texas Project Qualified Display Processing System.
May 22, 1986	Letter from applicant concerning Preservice Inspection of Component Supports; Relief Request.
May 23, 1986	Letter from applicant responding to NRC request for additional information regarding Regulatory Guide 1.97.
May 23, 1986	Letter from applicant concerning main steam isolation valve closure logic; NRC Question 440.57N.
May 23, 1986	Letter from applicant concerning momentary fuel assembly liftoff; supplemental information.
May 30, 1986	Letter from applicant responding to request for additional information regarding NRC Generic Letter 83-28.
May 30, 1986	Letter from applicant responding to "Status of Review of South Texas Project Information on Generic Letter 85-12."
May 30, 1986	Letter from applicant concerning revised schedule for response to 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."
June 4, 1986	Letter from applicant concerning Equipment Qualification Schedule - Revised Response to NRC Question 110.29.
June 4, 1986	Letter from applicant concerning 1985 Preoperational Radiological Environmental Monitoring Report.
June 9, 1986	Letter from applicant transmitting comments on the Safety Evaluation Report (SER), NUREG-0781.
June 11, 1986	Letter from applicant concerning maximum available fault current.
June 13, 1986	Partial Initial Decision (Operating License Phases II/III) issued by the Atomic Safety and Licensing Board.
June 13, 1886	Letter to applicant transmitting a copy of the June 10, 1986, ACRS Report for South Texas Project, Units 1 and 2.
June 13, 1986	Letter from applicant describing additional actions and programs initiated following the CAT inspection.
June 17, 1986	Letter from applicant transmitting a response to Safety Evaluation Report, NUREG-0781, Open Item 2, "Internal Missile Analysis."

June 17, 1986	Letter from applicant concerning Emergency Plan Procedures.
June 27, 1986	Letter from applicant transmitting additional information regarding thimble tube vibration.
June 27, 1986	Letter from applicant concerning the ACRS report issued by NRC on June 10, 1986.
June 27, 1986	Letter from applicant transmitting a response to SER Confirmatory Item 12; high head safety injection maximum discharge pressure.
June 30, 1986	Telephone conference call between NRC and applicant representatives regarding NRC review of fire hazards/safe shutdown. (Minutes issued June 30, 1986.)
July 7, 1986	Letter from applicant concerning subsidence monitoring.
July 9, 1986	Representatives from NRC, the applicant, Westinghouse, and Bechtel meet in Bethesda, Maryland, to prepare for the NRC staff audit. (Summary to be issued.)
July 15, 1986	Letter to Sorrento Electronics Division of GA Technologies withholding from public disclosure a report submitted on "Reliability Analysis Report for ESF Loan Sequencer, South Texas."
July 15, 1986	Letter to Westinghouse Electric Corporation withholding from public disclosure South Texas Project References CAW 85-04 and CAW 85-044.
July 15 & 16, 1986	Representatives from NRC, Sohar, Inc., the applicant, Westinghouse, and Bechtel meet in Monroeville, Pennsylvania, for the purpose of preparing for the NRC staff audit of Qualified Display Processing System. (Summary to be issued.)
July 18, 1986	Letter from applicant transmitting comments on the Draft Environmental Statement.
July 25, 1986	Letter from applicant concerning momentary fuel assembly liftoff; fuel assembly forces. (Proprietary withholding requested.)
July 28, 1986	Letter from applicant concerning main feedwater line break.
July 31, 1986	Letter from applicant concerning Fire Hazards Analysis Report Amendment 4.
August 29, 1986	Letter from applicant providing the test plan for vibration testing of the diesel fuel oil piping.

APPENDIX B

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---, NUREG/CR-2913, "Two Phase Jet Loads," January 1983.

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APPENDIX D

ACRONYMS AND INITIALISMS

ac	alternating current
AFWS	auxiliary feedwater system
AMSAC	ATWS mitigating system actuation circuitry
ATWS	anticipated transient(s) without scram
BE	best estimate
BMI	bottom-mounted instrumentation
BNL	Brookhaven National Laboratory
BTP	branch technical position
B&W	Babcock and Wilcox
CAT	construction appraisal team
CCW	component cooling water
CI	containment isolation
CVCS	chemical and volume control system
dc	direct current
EB	Engineering Branch
ECCS	emergency core cooling system
ECP	essential cooling pond
ECW	essential cooling water
EICSB	Electrical, Instrumentation and Control Systems Branch
EOP	emergency operations procedure
ERFDADS	emergency response facility data acquisition and display system
ERG	emergency response guideline
ESF	engineered safety feature
FOB	Facility Operations Branch
FSAR	Final Safety Analysis Report
GDC	general design criterion(a)
HHSI	high head safety injection
HL&P	Houston Lighting and Power Company
IVC	isolation valve cubicle
LMFW	loss of main feedwater
LOAC	loss of AC power
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
MCR	main cooling reservoir
msl	mean sea level

NPOD	Nuclear Plant Operations Department
NRC	U.S. Nuclear Regulatory Commission
PORC	Plant Operations Review Committee
PORV	power-operated relief valve
psig	pounds per square inch gauge
PWR	pressurized-water reactor
QDPS	qualified display processing system
RCP	reactor coolant pump
RCS	reactor coolant system
RG	regulatory guide
RTB	reactor trip breaker
RV	relief valve
SER	Safety Evaluation Report
SG	steam generator
SGTR	steam generator tube rupture
SOE	sequence of events
SRO	senior reactor operator
SRP	Standard Review Plan
STA	shunt trip attachment
STP	South Texas Project
SV	safety valve
STA	shift technical advisor
TMI-2	Three Mile Island Unit 2
UV	undervoltage
UVTA	undervoltage trip attachment
WOG	Westinghouse Owners Group

APPENDIX E

NRC STAFF CONTRIBUTORS AND CONSULTANTS

This Supplemental Safety Evaluation Report is the product of the NRC staff and its consultants. The NRC staff members listed below were principal contributors to this report.

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APPENDIX K

ACRS REPORT ON THE SOUTH TEXAS PROJECT, UNITS 1 AND 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 10, 1986

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON SOUTH TEXAS PROJECT, UNITS 1 AND 2

During its 314th meeting, June 5-7, 1986, the Advisory Committee on Reactor Safeguards reviewed the application of Houston Lighting and Power Company (HL&P), the Applicant, acting on behalf of itself and as agent for the City Public Service Board of San Antonio, Central Power and Light Company, and City of Austin for a license to operate the South Texas Project, Units 1 and 2. The ACRS commented on the construction permit application for the South Texas Project, Units 1 and 2 in a report dated September 19, 1975. The ACRS Subcommittee on the South Texas Project toured the facility on May 29, 1986 and met in Bay City, Texas on May 29 and 30, 1986 to discuss the application. During our review, we had the benefit of discussions with representatives and consultants of the Applicant, Westinghouse Electric Corporation, Bechtel Energy Corporation, and the NRC Staff. We also had the benefit of the documents referenced.

The site is located in south-central Matagorda County west of the Colorado River, 8 miles north-northwest of the town of Matagorda and about 89 miles southwest of Houston. The plant is located about 12 miles south-southwest of Bay City. Westinghouse Electric Corporation is the nuclear steam supply system and turbine-generator supplier for South Texas Project, Units 1 and 2. This Project makes use of identical four-loop Westinghouse pressurized water reactors and turbine generators. Unit 2 is similar to Unit 1 and is 600 feet away. This is the only U.S. plant using the RESAR-41 design. Although this design differs in some respects from other Westinghouse four-loop units in this country, it is quite similar to the Paluel plant in France, which is now in operation. Unit 1 is approximately 90 percent complete, and it is scheduled to load fuel in June 1987. Unit 2 is expected to follow about eighteen months later. The Applicant appears to have assembled a capable and experienced staff.

During our meeting, the NRC Staff identified a number of issues that must be resolved prior to the granting of an operating license. The residual heat removal pump is located inside containment. While this offers some advantages, it will be necessary that the pump be qualified for operation in an accident environment before this system can be judged acceptable. We wish to be kept informed.

We heard a report from a representative of the NRC's Region IV Office that construction quality and quality assurance effectiveness at the South Texas Project were satisfactory and that the attention being given by management to all aspects of the plant's readiness was commendable. However, the results of a recent Construction Appraisal Team inspection which are presently being considered may introduce items requiring attention.

In its report of September 19, 1975 on the construction permit application, the ACRS asked to be kept informed on the resolution of several items, including the location of the storage tanks for the diesel fuel. The diesel fuel storage tanks are located in separate rooms above the diesel generators. With this arrangement, a major concern is that a break in the piping between the storage tanks and the diesel generators will result in an uncontrolled discharge of fuel oil which may cause a fire. The ACRS recommends that the Applicant perform tests and take appropriate corrective measures to prevent failures in fuel oil piping and tubing by induced vibration resulting from extended operation of the diesel generators.

We believe that, subject to the resolution of open items identified by the NRC Staff and the items noted above, there is reasonable assurance that the South Texas Project, Units 1 and 2 can be operated at power levels up to 3800 Mwt without undue risk to the health and safety of the public.

Sincerely,

David A. Ward
Chairman

References:

1. Final Safety Analysis Report for South Texas Project, Units 1 and 2, Volumes 1-16, including Amendments 1-53
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of South Texas Project, Units 1 and 2," USNRC Report NUREG-0781 dated April 1986

APPENDIX L

ERRATA TO SOUTH TEXAS PROJECT, UNITS 1 AND 2, SAFETY EVALUATION REPORT

The errata on the South Texas Project SER is primarily based on comments received from the applicant dated June 9, 1986. The comments were extensive and covered the range of possibilities including typographical errors, editorial changes, and a few substantive corrections. The staff has reviewed the comments with the view that any comment which could improve the SER should be accommodated. However, where the comments are based on information not on the docket at the time the SER was issued, or representing a later submittal which has not been addressed in this supplement, no changes have been indicated. Significant changes which fall in this category will be evaluated in a future SER supplement which addresses the appropriate subject matter. For some comments, the staff did not agree with the requested change; therefore, the requested change does not appear in the errata.

South Texas Project SER

<table>
<tr><th>Page</th><th>Line/Item</th><th>Change</th></tr>
<tr><td>1-10</td><td>24</td><td>Change "three CHRS trains" to "two CHRS trains"</td></tr>
<tr><td>1-17</td><td>4</td><td>"10.4.9" to "10.4.9*"</td></tr>
<tr><td>1-18</td><td>7</td><td>Change "15.6.5" to "15.6.5*"</td></tr>
<tr><td>1-18</td><td>10</td><td>Change "7.2.2.6" to "7.2.2.4"</td></tr>
<tr><td>1-18</td><td>13</td><td>"15.6.5" to "15.6.5*"</td></tr>
<tr><td>1-18</td><td>14</td><td>"15.6.5" to "15.6.5*"</td></tr>
<tr><td>1-19</td><td>26-31</td><td>
<table>
<tr><th>Change</th><th>STP</th><th>CP</th><th>SNUPPS</th></tr>
<tr><td>Number of high pressure safety injection pumps</td><td>3</td><td>2</td><td>2</td></tr>
<tr><td>Number of intermediate safety injection pumps</td><td>3</td><td>2</td><td>2</td></tr>
<tr><td>Number of low pressure safety injection pumps</td><td>3</td><td>2</td><td>2</td></tr>
</table>
<table>
<tr><th>to:</th><th>STP</th><th>CP</th><th>SNUPPS</th></tr>
<tr><td>Number of high pressure pumps</td><td>2(1)</td><td>2</td><td>2</td></tr>
<tr><td>Number of intermediate pressure pumps</td><td>3</td><td>2</td><td>2</td></tr>
<tr><td>Number of low pressure pumps</td><td>3(2)</td><td>3(3)</td><td>2(3)</td></tr>
</table>
Add as footnotes:
(1) Charging only
(2) Dedicated to low head safety injection (SI) pumps; 3 separate dedicated residual heat removal (RHR) pumps also
(3) Low head SI/RHR function shared
</td></tr>
<tr><td>1-21</td><td>22</td><td>Change "9.5.1" to "9.5.1.7"</td></tr>
<tr><td>1-24</td><td>Add line 12</td><td>(35) Compliance with Generic Letter 85-12 (TMI Action Item II.K.3.5) RCP setpoint for small-break LOCAs (SER Section 15.6.5.1)</td></tr>
<tr><td>2-1</td><td>5</td><td>Change "12,300 acres (4978 hectares) to "12,200 acres (4937 hectares)"</td></tr>
<tr><td>2-1</td><td>13</td><td>Change "8.5 miles (13.7 km)" to "8.0 miles (12.9 km)"</td></tr>
</table>

2-1	16	Change "the plant" to "Unit 1"
2-1	32-34	Change "The applicant owns and controls all of the land and mineral rights within the designated exclusion area; no one resides within this area." to "No people reside within the exclusion area. The applicant has acquired all of the surface estate within the site boundary as well as most of the mineral interests within the site boundary. As the result of the acquisition of this surface estate and these mineral interests, the applicant has the authority required by 10 CFR 100 to determine all activities within the exclusion area."
2-3	38	Change "3057" to "FM 3057"
2-3	39	Change "2668" to "FM 2668"
2-3	40	Change "1095" to "FM 1095"
2-3	41-42	Change "Access to the plant, the visitors center, and the picnic area is by way of FM 521." to "Access to the plant and the visitors center is by way of FM 521."
2-5	30	Change "6.5" to "7.0"
2-10	34	Change "195" to "197"
2-14	3	Change "40,800" to "41,800"
2-14	33	Change "3580" to "3700"
2-15	3	Change "202,700" to "202,600"
2-15	8-9	Change "For the ECP, makeup water will also be provided from the Colorado River." to "For the ECP, makeup water will also be provided from the Colorado River via the MCR."
2-16	18	Add "Relocated" between "are" and "Little"
2-16	29	Add "Relocated" between "4.5-mi^2" and "Little"
2-16	44	Add "Relocated" between "4.5-mi^2" and "Little"
2-17	2	Add "Relocated" between "on" and "Little"
2-19	40	Delete "of piping" and add "from piping" between "seepage" and "through"
2-23	9-13	Change "Should any anomalies be detected, emergency procedures would be implemented to resolve any problems that could affect the stability of the MCR. These emergency procedures when implemented will require that watertight doors be normally

closed and that knockout panels be in place. These requirements will be addressed by the applicant in the MCR operating procedures."

to:

"Should anomalies which require the remedial action procedures to be implemented, administrative controls will be initiated to keep the watertight doors normally closed and knockout panels on the mechanical auxiliary building in place. These requirements will be addressed by the applicant in the MCR operating procedures."

2-24	33	Change "343.8" to "364"
2-26	2	Change "13.6" to "9.4 x 10^{-3}"
2-34	15	Change "FM 581" to "FM 521"
2-34	33	Delete "area"
2-35	23	Delete "widely"
2-37	29	Change "the vicinity of the site." to "the plant site."
2-38	35	Change "50 miles (80.5 km)" to "80 miles (128.7 km)"
2-45	28	Change "Figure 2.5.2-1A" to "Figure 2.5.1-1A"
2-45	39	Change "south of the site" to "south of the plant site"
2-45	39-40	Change "apparently extend to the surface." to "are assumed to extend to the surface."
2-50	33	Change "10^{-4}" to "10^{-2} to 10^{-3}"
2-51	22	Change "200" to "250"
2-51	28	Change "20 feet to 60 feet" to "50 feet to 80 feet"
2-54	19	Change "Appendix 2.5.8" to "Appendix 2.5.B"
2-55	38	Change "4- to 6-inch" to "up to 8-inch"
2-56	15	Change "Hendrow" to "Hendron"
2-57	30	Change "25" to "27"
2-57	34	Change "2.5.8.5.3-9" to "2.5.B.5.3-9"
2-60	2	Change "active" to "actual"

2-63 2-64	41 and 1	Change "An earth embankment 6200 feet in length was constructed water level." to "An earth embankment totalling 6,200 feet in length was constructed around the pond perimeter to provide an 8-foot-high freeboard for wave runup and to retain water during periods in which heavy precipitation would raise the water level."
2-64	1	Add "minimum" between "Normal" and operating"
2-69	41	Change "8.2 x 10^4 cm/sec" to 8.2 x 10^{-4} cm/sec"
2-73	11	Delete "south, west, and east sides of"
2-73	12	Change "were" to "are"
2-79	2	Change "Figure 2.2-1" to "Figure 2.2-2"
2-83	2	Change "Figure 2.4.3-30" to Figure 9.2.5-1"
2-110	15	Change "50-foot-diameter" to "51-foot-diameter"
2-110	20	Change "25 feet" to "15 feet"
3-15	11-13	Delete "All safety-related equipment near the eliminated break locations is environmentally qualified for the non-dynamic effects of a non-mechanistic pipe break with the greatest consequences on the equipment."
3-17	30	Add "and dynamic" between "static" and "resistance"
3-17	31	Add "and the wave propagation method," between "foundations" and "and"
3-17	35	Change "(accelerations)" to "(acceleration time history)"
3-17	37	Change "accelerations" to "acceleration time history"
3-19	18-22	Change "(1) Appropriate consideration for the most severe earthquake recorded for the site with an appropriate margin for intensity (GDC 2) and consideration of two levels of earthquakes (Appendix A, 10 CFR 100). The applicant has met this requirement by use of the acceptable seismic design parameter, per SRP Section 3.7.1." to "(1) Because of the minimum seismic design requirements promulgated in 10CFR Part 100 Appendix A,

		the seismic design is based on an equivalent earthquake intensity which is more conservative than the maximum historic earthquake determined for the tectonic province in which the plant is located. Therefore, the applicant has met the requirements set forth in the SRP Section 3.7.1."
4-1	13	Delete "or silver-indium-cadmium"
4-32	5	Change "indentical" to identical"
4-35	32	Delete "or emergencies"
5-5	39	Change "20 gpm" to "8 gpm"
5-12	17	Change "request" to "requests"
5-13	2-3	Delete "Flow or temperature devices are provided in the leak-off lines to indicate the source of leakage."
5-32	42	Change "panels" to "panel"
6-1	28	Change "2500-2600 ppm" to "2500-2700 ppm"
6-1	31	Change "34% to 36%" to "30% to 32%"
6-3	32	Change "3,410,000 ft^3" to "3.56×10^6 ft^3"
6-4	39	Change "0.14-ft^2" to "1.4 ft^2"
6-4	41	Change "0.14-ft^2" to "1.4 ft^2"
6-7	42-43	Change "Each steam generator compartment encloses a steam generator and a reactor coolant pump." to " Each steam generator compartment encloses two steam generators and two reactor coolant pumps."
6-9	18	Change "2900 gpm" to "1900 gpm"
6-11	19	Delete "an SI"
6-20	9	Change "RHR pumps" to "LHSI pumps"
6-28	30	Change "ANSI N509-1980" to "ANSI N509-1976"
6-28	31	Change "ANSI N510-1980" to "ANSI N510-1975"
6-29	22	Change "nonradioactive" to "filtered"
6-29	28	Add "a minimum of" between "supplies" and "2000 ft^3/min"
6-30	20	Change "vent stock" to "vent stack"

6-30	30	Add "or both" between "one" and "of"
6-35	22-23	Move "10,000" on line 22 to line 23
7-1	19	Change "Bechtel Corporation" to "Bechtel Energy Corporation"
7-7	4	Add "two out of three" between "from" and "emergency"
7-9	27-28	Change "These RTDs can be used as backup for T_{avg} and ΔT calculations." to "One of these RTDs can be used as backup for the cold leg temperature measurement."
7-9	34-35	Change "The staff concludes that the new design has better response time on temperature measurement and less maintenance problems and, therefore, is acceptable." to "The staff concludes that the new design meets present staff criteria and, therefore, is acceptable.
7-12	29-30	Change "(c) low compensated steamline pressure (2/3 in any steamline). (d) low pressurizer pressure (2/4)" to "(c) high-2 containment pressure (2/3) (d) low compensated steamline pressure (2/3 in any steamline)"
7-14	2	Change "mode II" to "mode III"
7-17	9	Change "(500,000 gallons)" to "(525,000 gallons)"
7-18	35	Change "maximum" to "minimum"
7-27	17	Change "...feedwater pump, which are located on the ASP." to "...feedwater pump, for which transfer switches are located on the ASP."
7-27	40-41	Change "(3) start/stop controls and transfer switches for the essential chillers located on each essential chiller local panel" to "(3) start/stop controls for the essential chillers located on each essential chiller local panel-transfer swtiches are on the transfer panels"
7-28	30-31	Change "The QDPS controls the flow into the steam generators through the AFW regulatory valves." to "The QDPS controls the flow into the steam generators

		through the AFW panel during safety injection. Also, position indication is provided by indicating lights on the main control panel and the ASP."
7-28	32-34	Add "are provided" after "limits"
7-31	13	Change "Qualfied" to "Qualified"
7-31	14-17	Change "The qualified display processing system (QDPS) is an integrated data acquisition and display system to cover postaccident monitoring, safety parameter display, inadequate core cooling monitoring, emergency response capability, and some limited safety-grade control functions." to "The qualified display processing system (QDPS) is an integrated data acquisition and display system to cover postaccident monitoring, inadequate core cooling monitoring emergency response capability, some limited safety-grade control functions, and inputs to the safety parameter display system.
7-31	41-42	Change "Three demultiplexers provide outputs to drive analog meters, recorders, computer, and annuciators." to "Three demultiplexers provide outputs to drive analog meters and recorders."
7-39	31	Change "open" to "auto"
7-48	27-29	Change "Since the steam generator PORV system is a Class 1E system, all portions of the steam generator PORV that could be exposed to an adverse environment are isolated in the isolation valve cubicle (IVC) on a loop-by-loop." to "Since the steam generator PORV system is a Class 1E system, all portions of the steam generator PORV that could be exposed to an adverse environment have been environmentally qualified. In addition, the PORV's are isolated in the isolation valve cubicle (IVC) on a loop-by-loop basis."
8-2	14	Change "these sets of towers" to "the middle and western towers"
8-3	1-3	Change "Current and voltage input to both primary and secondary protective relaying for each of the above circuits are respectively provided by separate sets

		of current transformers and voltage devices" to "Current and voltage input to both primary and secondary protective relaying for each of the above circuits are respectively provided by separate sets of current transformers and fused branch circuits from a set of voltage devices."
8-3	8	Insert "...open (negative terminals are normally tied) tie ..."
8-3	34-35	Change "(the two offsite power sources)" to "(two of the three preferred offsite power sources)"
8-4	26-31	Change "The applicant's steady-state (load flow) and transient stability analysis using criteria in FSAR Table 8.2.3 and results shown on FSAR Figures 8.2-6 through 8.2-12 demonstrate that outages of critical generators, faulting of critical buses, overload transmission circuits will not endanger the supply of offsite power to the ESF electrical system." to "The applicant's steady-state (load-flow) and transient stability analysis using criteria in the FSAR Table 8.2.3 and the results shown on FSAR Figures 8.2.6 through 8.2-12 demonstrate that outages of critical generators and faulting of critical buses will not endanger the supply of offsite power to the ESF electrical system, nor will they result in overloaded transmission circuits which would hinder the availability of the offsite power supply."
8-5	22	Change "48-V" to "480-V"
8-5	47	Change "150 hp" to "100 hp"
8-6	25	Change "be" to "the"
8-6	29-32	Change "Four of these UPSs are Class 1E. Two of these four Class 1E UPSs supply power to instrumentation Channels I and II and have their separate ac and dc power supplies from train A. The other two UPSs supply channels III and IV and have their power supplies from trains B and C; respectively." to "All of these UPSs are Class 1E. The UPSs supplying power to instrumentation Channels I and II have their separate ac and dc power supplies from train A. The other UPSs supplying channels III and IV

		have their power supplies from trains B and C, respectively."
8-14	29-31	Change "Centerline-to-centerline separation between adjacent electrical penetrations within a given train or channel is 4 feet." to "The vertical and horizontal separation distances between redundant separation groups are not less than the minimum acceptable separation distances of 3' horizontally and 5' vertically for general plant areas."
8-16	15	Change "also designed to comply with seismic Category I requirements." to "designed such that they will not degrade the ability of the safety systems to perform their safety function."
9-13	16	Delete "100% capacity"
9-13	16-17	Delete "(one for each unit with one on standby)"
9-13	46-47	Delete "by equipment in an adjacent building"
9-20	7	Change "by a grab sample" to "via the on-line determination with the PASS panel using ion chromatography with a grab sampled backup completed"
9-22	38	Add "and" between "charging" and "seal"
9-23	30	Delete "and is discussed in Section 12.3 of this SER"
9-30	5-6	Change "coiling" to "cooling"
9-30	8	Change "coiling" to "cooling"
9-32	13	Change "DBG" to "DGB"
9-57	6-10	Move this paragraph to the first paragraph of section 9.5.7 on page 9-62.
10-1	30	Change "two" to "three"
10-1	33	Change "digital" to "analog"
10-9	29-30	Change "the titanium tube ends and the aluminum bronze tube sheets" to "the water boxes."
10-11	17	Change "turbine building" to "outside"
11-1	38	Change "publishing" to "polishing"

11-2	27-29	Change "cartridge filter, an evaporator, a mixed-bed demineralizer, in series, before being collected in a waste management tank for sampling." to "waste monitor tank."
11-2	36	Change "4500 gpd" to "4650 gpd"
11-4	44	Change "dicussed" to "discussed"
11-7	14-15	Change "a mixing tank where chemicals can be added to neutralize the waste material." to "vendor supplied solidification or dewatering equipment to process liquid waste and spent resins, respectively."
11-8	13	Change "Redundant" to "Iodine, particulate, and noble gas"
12-2	21-22	Add "whenever possible" between "area" and "to"
13-1	18	Add "initial" between "the" and "fuel"
13-2	14	Add "the Manager of Nuclear Security" between "Training" and "and"
13-7	27	Change "LP-8.1" to "IP-8.1"
13-18	2	Delete "lead"
13-18	33	Delete "/acceptance"
13-18	35	Add "Acceptance testing is for non-safety related systems only."
13-20	42	Change "Plant Procedures Manual." to "Startup Administrative Instructions."
15-1	25	Change "turbine" to "reactor"
15-18	37	Add "(∼ 5%)" between "peaking" and "is"
15-18	38	Delete "(5%)"
17-1	33-35	Delete "(1) establishing QA indoctrination and training programs for personnel performing quality-affecting activities;"
17-1	35	Change "(2)" to "(1)"
17-1	35	Change "(3)" to "(2)"

17-1	37	Change "(4)" to "(3)"
17-1	38	Change "(5)" to "(4)"
17-1	39	Change "(6)" to "(5)"
17-1	40	Change "(7)" to "(6)"
17-1	41	Change "(8)" to "(7)"
17-2	9-11	Change "...a Technical Section responsible for technical support in the areas of engineering; and a Nuclear Training Section." to "...and a Technical Section responsible for technical support in the area of engineering."
Appendix G		
6	19-21	Change "(2) All safety-related equipment in the vicinity of Class 1 piping systems has been environmentally qualified for the nondynamic effects of a nonmechanistic pipe break with the greatest consequences on the equipment." to "(2) All safety-related equipment in the vicinity of Class 1 piping systems will continue to be environmentally qualified based on breaks postulated nonmechanistically at the previous AIB location."
9	3	Change "(2) All safety-related equipment in the vicinity of Class 2 and 3 piping systems has been environmentally qualified for the nondynamic effects of a nonmechanistic pipe break with the greatest consequences on the equipment." to "(2) All safety-related equipment in the vicinity of Class 2 and 3 piping systems will continue to be environmentally qualified based on breaks postulated nonmechanistically at the previous AIB location."
Appendix J		
16	1	Change "before" to "after"
25	Table 2	Change "1-15 years" to "1-5 years"

APPENDIX M

SAFETY EVALUATION REPORT
SOUTH TEXAS UNITS 1 AND 2
TORNADO MISSILE PROTECTION FOR
ISOLATION VALVE CUBICLE

TECHNICAL EVALUATION OF PROBABILISTIC RISK ASSESSMENT FOR TORNADO AND HURRICANE MISSILE HAZARD TO THE CONTAINMENT ISOLATION VALVE COMPARTMENT EQUIPMENT, SOUTH TEXAS PROJECT

U.S. DEPARTMENT OF COMMERCE LETTER
DATED DECEMBER 23, 1983

SAFETY EVALUATION REPORT
SOUTH TEXAS PROJECT UNITS 1 AND 2
TORNADO MISSILE PROTECTION FOR
ISOLATION VALVE CUBICLE
AUXILIARY SYSTEMS BRANCH

I. INTRODUCTION

Nuclear power plants must be designed to withstand the effects of tornado and high wind generated missiles so as not to impact the health and safety of the public in accordance with the requirements of General Design Criteria 2 and 4. The current licensing criteria governing tornado missile protection are contained in Standard Review Plan (SRP Section 3.5.1.4 and 3.5.2. These criteria generally specify that safety-related systems be provided positive tornado missile protection (barriers) from the maximum credible tornado threat. However, SRP Section 3.5.1.4 includes guidance on use of probabilistic risk assessment (PRA) methodology in lieu of the deterministic approach for assessing tornado missile protection. The acceptance criterion in this regard is similar to that identified in SRP Section 2.2.3 which deals with identification of design basis events using probabilistic methods. The tornado missile acceptance criterion is as follows:

"The probability of significant damage to structures, systems and components required to prevent a release of radioactivity in excess of 10 CFR Part 100 following a missile strike, assuming, loss of offsite power, shall be less than or equal to a median value of 10^{-7} per year or a mean value of 10^{-6} per year."

The following discussion of tornado missile protection is concerned with the isolation valve cubicles and the safety-related equipment within them.

The South Texas plant has four separate isolation valve cubicles (IVCs) each of which contains a portion of a main steam and a feedwater line. The main steam and feedwater isolation valves and main steam safety and relief valves associated with the steam and feedwater lines are also located within the cubicles. Each IVC is missile protected from all sides by heavy concrete walls. The tops of the cubicles however are open. A tornado missile(s) could enter one or all of the cubicles through the open top and damage the components therein.

The applicants elected to demonstrate compliance with the tornado missile protection criterion for the IVCs by PRA methodology rather than provide positive protection for the roof opening. The applicants provided a detailed PRA in a submittal dated September 13, 1983. Additional information to support the PRA was provided in submittals dated November 14 and December 20, 1983.

Due to the specialized nature of the study, we have contracted with the National Bureau of Standards (NBS) to assist in the review of the applicants' analysis. NBS provided a technical evaluation report (TER) regarding the probability of a tornado missile strike upon the IVCs. Concerns which were identified during our review were satisfactorily resolved by the applicants' response dated December 20, 1983 as indicated in our consultants TER supplement dated December 23, 1983. In this supplement the consultant also addressed concerns associated with missiles generated by non-tornadic and non-hurricane winds. The consultant's TER as supplemented forms a part of our SER.

II. EVALUATION

As previously stated, the South Texas Plant is designed with four separate IVCs, each of which is missile protected from all sides by heavy concrete walls. Each cubicle is completely protected except for the roof which is open. The height of the IVC walls is 55 feet above plant grade.

The applicants' PRA considered all of the SRP Section 3.5.1.4, November 24, 1975 Missile Spectrum A as potential missiles including the utility pole and the automobile. Revision 2 of the SRP however, allows the exclusion of the utility pole and the car at elevations up to 30 feet above all grade levels within 1/2 mile of the facility structures under review. As the height of the IVC wall is 55 feet above plant grade the missiles which we consider to apply from Missile Spectrum A are the wood plank, the steel rod and the steel pipes. Our examination of elevated areas within 1/2 mile of the facility structures disclosed only the dike area around the ultimate heat sink which could be considered as a possible launch point for the automobile or the utility pole. The applicants have assured us that there will be no utility pole storage along the dike area. Additionally, the only vehicular traffic along the dike would be transient in nature in order to conduct inspection, and this traffic will be controlled.

In order for a missile to strike any of the components in a given IVC, it must approach the roof opening at a steep angle, within a given solid angle. The roof opening of each IVC is approximately 745 square feet thus presenting a relatively small target. Additionally, the safety-related target areas within the IVCs are much smaller than the IVC open roof areas. The fact that there are four separate cubicles substantially decreases the probability of single missile being capable of damaging more than the components in one cubicle. Multiple missiles however, could enter separate cubicles. We consider this a low probability event, as discussed further below.

Rather than utilize a deterministic argument, as discussed above, the applicants chose to provide a PRA evaluation. The applicants' PRA was provided in their submittal of September 13, 1983. We and our consultant have reviewed this submittal. The review resulted in additional concerns which were identified to the applicants. The applicants provided responses to those concerns in a submittal dated December 20, 1983.

Our consultant's evaluation of the applicants' PRA considered the validity and conservatism of the approach, assumptions, and data used in the applicants' analysis to establish the probability of tornado and hurricane-borne missile damage to the IVC equipment. Also included in the evaluation is an assessment of the correctness of the results obtained in the study.

We have reviewed our consultant's TER and his supplement thereto contained in letter dated December 23, 1983 which resolved the open items identified in the TER.

We concur with the findings and resulting estimate of the probability of damage to essential equipment in IVC of 3×10^{-9}. We further agree that this value is correct to within at least one order of magnitude uncertainty. Therefore additional positive tornado missile protection need not be provided for the IVCs since the probability of exceeding 10 CFR 100 dose criteria due to tornado missiles is less than the 10^{-7} per year acceptance criterion.

Based on the above, we conclude that the applicants have satisfactorily demonstrated compliance with General Design Criteria 2 and 4 with respect to tornado missile protection for the IVCs. The design of the IVCs is therefore acceptable without the addition of further protection for the roof area.

TECHNICAL EVALUATION OF
PROBABILISTIC RISK ASSESSMENT FOR TORNADO AND HURRICANE MISSILE HAZARD TO THE CONTAINMENT ISOLATION VALVE COMPARTMENT EQUIPMENT
SOUTH TEXAS PROJECT

Emil Simiu

INTRODUCTION

The objective of this evaluation is to assess the validity and conservatism of the approach, assumptions, and data used in the Bechtel South Texas Study 14926-001 (August 1983 and November 1983)[a] to estimate the probability of tornado- and hurricane-borne missile damage to the containment isolation valve compartment (IVC) equipment. Also included in the evaluation is an assessment of the correctness of the results obtained in the study.

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a/ The Bechtel South Texas Study consists of two distinct parts. The first part deals with tornado-borne missile hazards. The second part deals with hurricane-borne missile hazards. These parts are referred to in the following sections as the Tornado Study and the Hurricane Study, respectively.

1. TORNADO STUDY

1.1 Assumptions Used in Tornado Study

The following main assumptions are used in the Tornado Study:

1. The distribution of the tornado occurrence rate, ν, is consistent with the National Weather Service record of tornado strikes for the site region[a] between 1953 and 1982 (Ref. 1), and is modeled by a lognormal distribution.
2. The distribution of the path area, a, is consistent with National Weather Service data for tornado strikes in Texas between 1953 and 1982 (Ref. 1), and is modeled by a lognormal distribution.
3. The joint distribution of the tornado path area, a, and the Fujita scale, F, is consistent with the National Weather Service nationwide record of tornadoes between 1953 and 1982 (Ref. 1), and is modeled by a continuous bivariate lognormal distribution.
4. The parameters CA/m (C = aerodynamic coefficient, A = area exposed to wind, and m = mass of missile)., which characterize the behavior of the missile in flight in any given wind field and for any given initial conditions, have a single set of values corresponding to a conventional "standard missile" (Table D.4 of Tornado Study). The values are based on the data on aerodynamic coefficients obtained from Ref. 2.
5. The median surface density of potential missiles over the entire missile origin zone (~ 0.9 square mile) is about 1 missile/(65 ft x 65 ft) [~ 6,600 missiles/sq. mile], based on an Electric Power Research Institute (EPRI) survey of seven nuclear power plant sites (Ref. 3). The surface

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[a] The site region consists of the following countries: Matagorda, Brazoria, Fort Bend, Wharton, Jackson, and Calhoun.

density has a lognormal distribution such that in 95 percent of the cases it is less than about 1 missile/(40 ft x 40 ft) [~ 17,000 missiles/sq. mile] in the area that might affect the target.

6. Fifty percent of the potential missiles have elevations uniformly distributed between grade level and 20 ft above grade, while the remaining 50 percent are at grade level.
7. Ninety percent of the elevated missiles are restrained.
8. Sheltering by other structures and by the roof at top of the IVC is neglected.
9. A missile strike on any portion of the IVC top results in total failure, i.e., no allowance is made for partial damage or for redundancy of components.
10. No allowance is made for the fact that safety related target areas are less than area of IVC top.

Additional assumptions, some of them tacit, are used in the Appendices to the Tornado Study. These assumptions are pointed out and commented upon subsequently in the evaluation.

1.2 Data Used in Tornado Study

Data used in the Tornado Study include:

1. Number of tornadoes recorded in 1953-1982 in the site region defined in Section 1.1 herein. These data are taken from Ref. 1.
2. Tornado path areas for tornadoes recorded in Texas in 1953-1982 (from Ref. 1).
3. Tornado path areas and Fujita scale classification for tornadoes recorded nationwide in 1953-1982 (from Ref. 1).

4. Area and population density data for the six counties listed in item 1 of Section 1.1, and U.S. population density data in each of the years 1950-1979.
5. Survey data on distributions of missiles by number and length (from Ref. 3).
6. Aerodynamic data for various missiles listed in Table D.3 of Tornado Study (based in part on Ref. 2).

1.3 Mathematical Approaches Used in Tornado Study

The fundamental approach to estimating the probability, P_T, of damage due to tornado missile hits is the estimate the factors P_o ad P_H in the expression

$$P_T = P_o \cdot P_H \tag{1}$$

where P_o = probability per year that a tornado strikes the plant site,

P_H = probability of hitting the target assuming that a tornado occurs at the site.

In the Tornado Study an attempt is made to provide not only point estimates of P_T, but confidence limits for those estimates as well.

1.3.1 Probabilities of Tornado Occurrence at the Site, P_o. Probabilities of occurrence at the site of a tornado with path area, a, are estimated as

$$P_o(a) = \frac{\nu a}{S} \tag{2}$$

where ν = annual frequency of a tornado occurrence in the geographical area S within which it may be assumed that tornado rates of occurrence are uniform. (In the Tornado Study, S = 10,000 sq. miles.)

As mentioned in items 1 and 2 of Section 1.1 herein, it is assumed that the distributions of the annual occurrence rate, ν, and of the tornado path area, a, are lognormal. These assumptions may be expected to provide conservative estimates as far as the 95-th percentile of the probability of damage are concerned. It is this percentile that is used in the Tornado Study to assess the risk of damage to the IVC equipment (see Table III, p. 8 of the Tornado Study). For this reason it is the reviewer's opinion that the results obtained in Appendix D of the Tornado Study concerning the modeling of ν and a are acceptable for the purposes of this probabilistic risk assessment.

It is noted that the estimated annual rate of tornado occurrence is adjusted upward to account for possible tornado underreporting due to low population density. This adjustment, which in the case of this project is a minor one[a], is carried out in accordance with the procedure developed in Appendix B of the Tornado Study. This procedure is based on the assumption that there is a statistical correlation between the number of tornadoes reported nationwide and the population density in the U.S.A. during each of the years 1950-1979. In the reviewer's opinion this procedure is reasonable.

1.3.2 Probabilities of Hitting the Target Assuming that a Tornado Occurs at the Site. The probability of hitting a target with area A given that a tornado with intensity F on the Fujita scale occurs at the site is expressed in the Tornado Study as

$$P_H(F) = n_p A \eta(F) \psi(z,F) \quad (3)$$

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[a] See Eq. D.36, p. D-12 of the Tornado Study.

where n_p = number of potential missiles per unit area (see item 5, Section 1.1 herein)

A = area of target

$\eta(F)$ = probability that a missile swept by a tornado with intensity F will become airborne

$\psi(z,F)$ = probability that a horizontal unit area with elevation z will be hit by a missile borne by a tornado with intensity F, given that the density of missiles at the site swept by the tornado is one per unit of horizontal area.

1.3.3 Probability of Injection $\eta(F)$. Probabilities of injection are estimated in accordance with models and calculations presented in Appendix C of the Tornado Study. Inherent in these estimates are two assumptions, both of which are conservative. First, it is assumed that the speed of the tornado, w, is constant throughout the tornado width. Second, it is assumed that the random angle θ between the drag force vector and the vertical is uniformly distributed between the values $\theta = 0$ and $\theta = \pi$. In reality, just prior to the take-off, the drag force vector and the wind velocity vector coincide, i.e., $\theta = 0$, so that in most cases this second assumption strongly overestimates the vertical component of the drag force at the take-off. For example, to the case $\theta = \pi/2$ there corresponds in the Tornado Study the assumption that the direction of the tornado wind speed, w, is vertical. In reality, there are strong reasons to believe that the vertical component of tornado wind speeds is always significantly less than w.

The probability of injection depends upon the assumed values of the drag and lift restraint coefficients, K_D and K_L. There are few data on these values, which must therefore be assumed on the basis of judgment. The uniform distributions $0 < K_D < 5$ and $1 < K_L < 5$ are used except as noted below. The corresponding injection probabilities are listed in Tables C-11 and C-13 and Tables D-17 and D-16 of the Tornado Study. For the restrained elevated missiles the lower limit of the drag restraint coefficient is $K_D = 1$, rather than $K_D = 0$, and the corresponding injection probabilities are listed in Tables C-15 and D-18 of the Tornado Study.

In the reviewer's opinion, the assumptions regarding the restraint coefficients used in the Tornado Study are reasonable. However, as mentioned in Section 3 of this review, these assumptions are difficult to evaluate owing to the lack of sufficient and clearly interpretable experimental data.

1.3.4 Height Distribution of Airborne Missiles, $\psi(z,F)$. The most elaborate part of the Tornado Study deals with the derivation of the function $\psi(z,F)$. The approach used to derive this function is now briefly described.

First, it is postulated, as in the EPRI report NP-769 [3], that the movement of a tornado missile can be viewed as a Markov chain[a]. This postulate is justified, as in Ref. 3, by the assumption that the missile can be viewed as undergoing purely random tumbling, so that the aerodynamic force it will experience at any one point depends on the random position that the missile has at that point, rather than on the previous geometric attitudes of the missile.

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[a] A Markov chain is a process in which the probability of transition from one point to another depends only on the coordinates of these points and on the state of the system at the initial point, i.e., the probability is independent of the previous history of the system.

The extent to which this justification is acceptable is difficult to assess. In Ref. 2 tornado-borne missiles were treated as six-degree-of-freedom systems, with tumbling determined by mechanical relations, rather than occurring randomly. However, that model is not necessarily superior to the Markov chain representation because it does not appear to account for Magnus effects, random disturbances due to turbulence inherent in the flow or induced by the missile, and random initial missile attitudes. An alternative model is one in which the missile is viewed as a material point subjected to drag, and the drag coefficient is assumed to be uniform and equal to an average of the aerodynamic coefficients corresponding to all possible attitudes. This model is also fundamentally unsatisfactory. In the absence of better practical choices, this reviewer is inclined to view the acceptance of the Markov chain model as a reasonable option.

Once the Markov chain model for the missile motion is postulated, the second step is to define a probability density function, $G(\overline{r}_0, \overline{v}_0, t_0, \overline{r-r_0}, \overline{v-v_0}, t-t_0, F, \gamma)$, such that

$$dP = G dx dy dz \, dv_x \, dv_y \, dv_z \tag{4}$$

where x, y, z = coordinates in space,

v_x, v_y, v_z = missile velocity components,

and dP = probability that, given the occurrence of a tornado with intensity F and characteristics γ, a missile that became airborne at movement t_0 hits the volume $dxdydz$ around point $\overline{r}$ during a unit time interval at moment t, with a velocity between $\overline{v}$ and $\overline{v} + d\overline{v}$ ($\overline{r} = x\overline{i} + y\overline{j} + z\overline{k}$; $\overline{v} = v_x\overline{i} + v_y\overline{j} + v_z\overline{k}$, where $\overline{i}, \overline{j}, \overline{k}$ = unit orthogonal vectors). The function G is referred to as the original (fundamental) Green's function of the problem. Modified Green's functions

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can be derived by integrations and/or averaging of the original Green's functions. Modified functions correspond to (1) the probability that, given the occurrence of a tornado with intensity F and a set of characteristics γ, the missile will hit the volume dxdydz around point $\overline{r}$ during a unit time interval at moment t with any velocity, (2) a similar probability for the case where the hit occurs over a unit area with orientation $\overline{\Omega}$, (3) a similar probability averaged over all possible tornadoes having intensity F and area a.

By applying the Fokker-Planck equation to the function G and integrating and averaging the results as done to obtain the probabilities (2) and (3) above, the Tornado Study derives closed form relations for the function $\psi(z,F)$, i.e., for the probability that the horizontal unit area with elevation z will be hit by missiles borne by tornadoes with intensity F that sweep an area for which the surface density of potential missiles is 1 missile/ft^2.

The reviewer has verified the derivations leading to the expressions for $\psi(z,F)$ and has concluded, to the best of his ability, that they are correct. He believes that the statistical mechanics approach on which these expressions are based can provide useful insights into the question of tornado-borne missile damage, and acceptable order of magnitude estimates of probabilities of tornado-borne missile hits.

2. HURRICANE STUDY

The probability, P_T, of damage due to hurricane-borne missile hits is written as

$$P_T = \int_0^\infty f(w)\ P_H(w)\ dw \qquad (5)$$

where f(w) = probability density function of hurricane wind speeds, w, at the site, and $P_H(w)$ = probability of one or more hurricane-borne missiles striking the top of any one of four IVC compartments, given the occurrence of a hurricane with speed w.

The probability $P_H(w)$ is modeled in a manner identical to that used in the Tornado Study. (Note that in the Tornado Study the dependence of the probability P_H upon Fujita scale F is converted into a dependence upon wind speed w, since to each intensity on the Fujita scale there corresponds a range of speed w.) Calculated values of $P_H(w)$ for restrained and unrestrained missiles are listed for various wind speeds w in Tables A1.a and A1.b of the Hurricane Study. The probability density function, f(w), is assumed to be consistent with the values of Tables I and B.2 of the Hurricane Study.

2.1 Probability P_H

To the extent that the model of the probability P_H is acceptable for tornado winds, it can also be considered acceptable for hurricanes, with the following qualification. The height distribution of airborne missiles, $\psi(z,w)$ (discussed in Section 1.3.4 of this report) is dependent upon (1) the wind field within the storms being considered, and (2) the probability distribution of the directions of translation of the storms. In the case of hurricanes, both these factors differ from their tornado counterparts. In particular, hurricanes tend to exhibit preferred translation velocities, whereas the model upon which $\psi(z,w)$ is based assumes uniform directional distribution of those velocities. Note, however, that $\psi(z,w)$, as tabulated in Tables A1.a and A1.b of the Hurricane Study, is of the order of 0.20 or higher, and that, by definition, $\psi(z,w) < 1$. Therefore, accounting for differences between the respective wind fields and directions of translation could increase $\psi(z,w)$ by a factor of the order of at most five. (Note that $\psi(z,w)$ could also decrease.) Since the estimated median value of the probability of damage to the IVC from hurricane-generated missiles is 1.2×10^{-10} per year (Table III of Hurricane Study), a function $\psi(z,w)$ estimated on the basis of assumptions different from those used in the Tornado Study would not alter in this case the conclusion that the risk of damage to the IVC is acceptable. However, this statement is correct only to the extent that the assumption used in the Hurricane Study with respect to the probability density function of hurricane wind speeds is acceptable. This assumption is examined below.

2.2 Probability Distribution of Hurricane Wind Speeds

The probability density function of the hurricane wind speeds used in the Hurricane Study is consistent with the values listed in Table I therein.

Reference 5 lists parameters of the Weibull distributions that best fit hurricane wind speeds generated by Monte Carlo simulation in accordance with the procedure described in Ref. 4. Table 2.2.1 shows the values of these parameters for mileposts 300 and 350 (see Fig. A-1 of Hurricane Study).

Table 2.2.1 Parameters of Weibull Distribution of Hurricane Wind Speeds [5]

Milepost	300			350		
	μ	σ	γ	μ	σ	γ
	-1,328.5	1,344.0	30	-435.9	453.9	11

Note: μ and σ correspond to speeds averaged over 1 minute and are expressed in knots.

Wind speeds corresponding to various probabilities of exceedence based on the parameters of Table 2.2.1 are shown in Table 2.2.2, which also includes the values estimated in the Hurricane Study.

Table 2.2.2 Wind Speeds Corresponding to Various Probabilities of Exceedence[a]

Probability of Exceedence Per Year	Table I, Hurricane Study (at Site)	Based on Ref. 5	
		Milepost 300	Milepost 350
5.4×10^{-3}	112	105	105
6.1×10^{-4}	134	124	124
2.9×10^{-5}	157	144	145
6.0×10^{-7}	179	161	164

[a] Wind speeds in mph averaged over 1 minute.

It is seen from Table 2.2.2 that the hurricane wind speed probabilities assumed in the Hurricane Study are comparable to those based on the Weibull parameters of Ref. 5. To the extent, then, that the estimates of hurricane wind speeds based on Ref. 4 (or, equivalently on Ref. 5) are acceptable, it follows from

Table 2.2.2 that the probability distribution f(w) used in the Hurricane Study (see Eq. 5 of this review) is slightly conservative.

As in the case of tornadoes, estimates of extreme hurricane wind speeds corresponding to very small exceedence probabilities are uncertain. However, the reviewer feels that estimates based on Ref. 4 are comparable in terms of reliability to those obtained in the current state-of-the-art for tornado winds.

3. ASSESSMENT OF BECHTEL SOUTH TEXAS STUDY[a] - SUMMARY

In this reviewer's opinion, the approach used in the Bechtel South Texas Study is physically and mathematically acceptable for engineering purposes. The assumptions used therein appear to be conservative, although reservations are expressed concerning:

1. The feasibility of estimating tornado and hurricane wind probabilities corresponding to mean recurrence intervals of the order of tens of thousands of years or more from data recorded over 30 years or so. It is the reviewer's opinion that no literal meaning should be attached to such estimates as carried out by current methods, including the methods used in the Bechtel South Texas Study. However, these estimates have some validity in a relative sense, i.e., they are capable of providing comparative engineering assessments of hazards associated with strong winds. As such, the reviewer feels that the estimates obtained in the Bechtel South Texas Study are reasonable and acceptable given the present state-of-the-art.
2. The surface density of the potential missiles is assumed to have a median of 1 missile/65 ft x 65 ft (or about 6,600 missiles/sq. mile). Although this assumption is consistent with data published in Ref. 3, it is possible that the actual surface density of potential missiles will be higher. In order to minimize this probability, the Regulatory Staff should, in the reviewer's opinion, be satisfied that structures located within, say, 200 m of the IVC are capable of withstanding the Design Basis Tornado without loss of integrity.

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[a] See footnote in the Introduction to this review

3. The possibility that missiles lighter than those assumed in the Bechtel South Texas Study (e.g., two by fours) might be available, become airborne, and damage the IVC.
4. The possibility that the assumptions used in the Bechtel South Texas Study concerning the missile injection do not correspond sufficiently closely to the unknown physical reality. As indicated in Section 1.3.3, although they appear reasonable, these assumptions are difficult to evaluate owing to the lack of sufficient and clearly interpretable data. The reviewer believes that the numerical results based on these assumptions are credible (e.g., it appears credible that the median probability of injection of standard unrestrained missiles in storms with maximum winds of less than 183 mph is zero - see Table A1.b of Hurricane Study). However, this is to some degree a subjective view.

With the above reservations, it is the reviewer's opinion that the conclusions of the Bechtel South Texas Study concerning the tornado and hurricane missile hazard to the IVC equipment are acceptable.

REFERENCES

1. U.S. Tornado Breakdown by Countries 1953-1982, U.S. Department of Commerce, National Severe Storms Forecast Center, Federal Building, 601 E. 12th Street, Kansas City, Missouri 64106.
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4. Batts, M.E. et al., "Hurricane Wind Speeds in the United States," Journal of the Struct. Division, ASCE, October 1980, pp. 2001-2016.
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UNITED STATES DEPARTMENT OF COMMERCE
National Bureau of Standards
Washington, D.C. 20234

December 23, 1983

Mr. B. K. Singh
Mail Stop P-1022
Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Singh:

Subject: Evaluation of Supplemental Information Pertaining to Probabilistic Risk Assessment (PRA) for Tornado and Hurricane Missile Hazard to the Containment Isolation Valve Compartment Equipment, South Texas Project

The report "Technical Evaluation of Probabilistic Risk Assessment for Tornado and Hurricane Missile Hazard to the Containment Isolation Valve Compartment Equipment, South Texas Project," was based on documents made available to the National Bureau of Standards before December 12, 1983. Following that date, I have received from NRC: (1) a copy of questions addressed by NRC to Bechtel concerning the subject PRA, and of responses from Bechtel to these questions (attached herewith), and (2) a copy of NRC memorandum Docket Nos. 50-498/499 dated December 13, 1983, to George W. Knighton and Olan D. Parr from William P. Gammill, all of NRC, on the Meteorology Review of Probabilistic Evaluation of Isolation Valve Cubicled Roof Design for South Texas.

NRC Questions to Bechtel

In my opinion, given the present state of the art, Bechtel's responses to the NRC questions satisfactorily answer concerns that may be raised with respect to:

1. the missile surface density assumed in the Bechtel South Texas Study

2. the possibility that missiles lighter than those assumed in the Bechtel South Texas Study might be available and damage the Isolation Valve Compartment, and

3. the possibility that Bechtel's estimates of injection probability may be somewhat low. In the event that the threshold hurricane wind speed for missile injection is reduced to 165 mph, the median probability of hitting the target will increase by about one order of magnitude if 10 percent of the missiles are unrestrained and by about two orders of magnitude if all missiles are unrestrained.

NRC Meteorology Review

With respect to the NRC Meteorology Review, I note that most of the difference between the NRC and the Bechtel estimate of probability that a tornado strikes the site in any one year (1.7×10^{-4} per year and 1.17×10^{-5} per year) is due to the difference between the respective estimates the mean tornado path area (0.206 square miles by NRC; about 0.049 square miles, to which there corresponds a median of 0.022 square miles, by Bechtel).

The NRC mean tornado path area was estimated from a sample of 250 tornadoes (period of occurrence 1954 - 1981) reported in the two-degree latitude-longitude square (excluding overwater area) containing the plant site. On the other hand, the Bechtel estimate was based on a sample of 2,730 tornadoes reported in 1953 - 1982 throughout the state of Texas. I believe that tornadoes in south Texas and in the other areas of Texas do indeed not constitute a meteorologically homogeneous set and, therefore, that the NRC estimate is more credible. This would increase the rate of occurrence of tornadoes at the site and, hence, the nominal probability of damage to IVC from tornado-generated missiles by about one order of magnitude. Thus, the estimated median of probability of damage to IVC from tornado-generated missiles would be about 3×10^{-9}, rather than 2×10^{-10} as indicated in the Bechtel study.

The NRC memorandum mentioned above notes that the Bechtel estimate of hurricane wind speeds appears reasonable. This is also my view. Thus, the NRC memorandum would not entail a modification of the estimated median value of the nominal probability of damage to IVC by hurricane-borne missiles put forth by Bechtel (i.e., 1.2×10^{-10} per year). However, the NRC memorandum points out that winds other than hurricanes and tornadoes were not considered in the Bechtel study.

Calculations conducted at NBS and reported in "Hurricane Wind Speeds in the United States" by M.E. Batts, et al. (Journal of the Structural Division, ASCE, October 1980, pp. 2001-2016) indicate that the effect of nonhurricane and nontornadic winds on the probability of occurrence of winds of any given speed is perceptible only for speeds corresponding to mean recurrence intervals of about 25 to 50 years at most. Therefore, it is my opinion that failure to consider winds other than hurricanes and tornadoes does not affect the calculations of the Bechtel study.

The estimated median value of the nominal probability of damage to the IVC by tornado or hurricane-borne missiles would then be about $3 \times 10^{-9} + 2 \times 10^{-10} \approx 3 \times 10^{-9}$. Given the various uncertainties inherent in the Bechtel estimates (including uncertainties with respect to the probability of injection as reflected in the Bechtel answer to question #3 by NRC), it is my opinion that the estimated probability of damage to the IVC of 3×10^{-9} is correct to within one or two orders of magnitude.

Sincerely,

Emil Simiu
Research Structural Engineer
Structural Engineering Group
Structures Division
Center for Building Technology, NEL

cc: S. Boyd
L. Rubenstein
B. Grenier

APPENDIX N

EG&G TECHNICAL EVALUATION REPORT FOR SOUTH TEXAS UNITS 1 AND 2 CONFORMANCE TO GENERIC LETTER 83-28 ITEM 2.1 (PART 1)

EGG-NTA-7237

CONFORMANCE TO GENERIC LETTER 83-28 ITEM 2.1 (PART 1) EQUIPMENT CLASSIFICATION (RTS COMPONENTS)

SOUTH TEXAS 1 AND 2
VIRGIL C. SUMMER 1
TROJAN
YANKEE ROWE

R. Haroldsen

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

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under DOE Contract No. DE-AC07-76ID01570
FIN Nos. D6001 and D6002

ABSTRACT

This EG&G Idaho, Inc. report provides a review of the submittals from selected operating Pressurized Water Reactor (PWR) plants for conformance to Generic Letter 83-28, Item 2.1 (Part 1). The following plants are included in this review.

Plant Name	Docket Number	TAC Number
South Texas 1	50-498	--
South Texas 2	50-499	--
Summer 1	50-395	52885
Trojan	50-344	52890
Yankee Rowe	50-29	52895

FOREWORD

This report is supplied as part of the program for evaluating licensee/applicant conformance to Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events." This work is being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of PWR Licensing-A, by the EG&G Idaho, Inc.

The U.S. Nuclear Regulatory Commission funded this work under the authorization B&R 20-19-10-11-3 and 20-19-40-41-3, FIN Nos. D6001 and D6002.

CONTENTS

1. INTRODUCTION AND SUMMARY

On February 25, 1983, both of the scram circuit breakers at Unit 1 of the Salem Nuclear Power Plant failed to open upon an automatic reactor trip signal from the reactor protection system. This incident was terminated manually by the operator about 30 seconds after the initiation of the automatic trip signal. The failure of the circuit breakers was determined to be related to the sticking of the undervoltage trip attachment. Prior to this incident, on February 22, 1983, an automatic trip signal was generated at Unit 1 of the Salem Nuclear Power Plant based on steam generator low-low level during plant startup. In this case, the reactor was tripped manually by the operator almost coincidentally with the automatic trip.

Following these incidents, on February 28, 1983, the NRC Executive Director of Operations (EDO), directed the staff to investigate and report on the generic implications of these occurrences at Unit 1 of the Salem Nuclear Power Plant. The results of the staff's inquiry into the generic implications of the Salem Unit 1 incidents are reported in NUREG-1000, "Generic Implications of the ATWS Events at the Salem Nuclear Power Plant."[1] As a result of this investigation, the Commission (NRC) requested (by Generic Letter 83-28, dated July 8, 1983)[2] all licensees of operating reactors, applicants for an operating license, and holders of construction permits to respond to generic issues raised by the analyses of these two ATWS events.

This report is an evaluation of the responses submitted from a group of similar pressurized water reactors for Item 2.1 (Part 1) of Generic Letter 83-28.

The results of the reviews of several plant responses are reported on in this document to enhance review efficiency. The specific plants reviewed in this report were selected based on the similarity of plant design and convenience of review. The actual documents which were reviewed for each evaluation are listed at the end of each plant evaluation. The generic documents referenced in this report are listed at the end of the report.

Part 1 of Item 2.1 of Generic Letter 83-28 requires the licensee or applicant to confirm that all reactor trip system components are identified, classified, and treated as safety-related as indicated in the following statement:

> Licensees and applicants shall confirm that all components whose functioning is required to trip the reactor are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders, and parts replacement.

2. PLANT RESPONSE EVALUATIONS

2.1 South Texas 1 and 2, 50-498/499 (OL Plants)

The applicant for South Texas Units 1 and 2 (Houston Lighting and Power Co.) provided a response to Item 2.1 (Part 1) in a submittal dated June 28, 1985. The submittal states that the applicant had conducted a review of all components whose function is required to trip the reactor. These components were verified to be properly classified. The classification is designated in design documents and the plant Q-List. Maintenance, work orders and parts replacement require identification of safety classification prior to approval.

2.2 Conclusion

Based on the review of the applicant's submittal, we find that the applicant's response confirms that the components required to trip the reactor are identified as safety-related, and that documents used to initiate design changes, maintenance, or procurement require identification of safety-related components. The licensee's responses, therefore, meet the requirements of Item 2.1 (Part 1) of Generic Letter 83-28, and is acceptable.

Reference

1. Letter, J. H. Goldberg, Houston Lighting and Power Co., to H. L. Thompson, Jr., NRC June 28, 1985.

2.3 Virgil C. Summer 1, 50-395, TAC No. 52885

The licensee for the Virgil C. Summer 1 Nuclear Plant (South Carolina Electric and Gas Co.,) provided a response to Item 2.1 (Part 1) in a submittal dated November 4, 1983. The submittal states that the components of the reactor trip system have been reviewed and verified to be properly classified. In addition the design documents such as drawings, specifications and bills of material are issued with safety-related designations and replacement parts are procured via safety-related procurement documents. The licensee at the time of the submittal was integrating the equipment classification lists into a Computerized History and Maintenance Planning System. This system was scheduled for completion in January 1984.

2.4 Conclusion

Based on the review of the licensee's submittal, we find that the licensee's response confirms that the components required to trip the reactor are identified as safety-related, and that documents used to initiate design, maintenance, or procurement require identification of safety-related components. The licensee's response, therefore, meets the requirements of Item 2.1 (Part 1) of Generic Letter 83-28, and is acceptable.

References

1. Letter, O. W. Dixon, Jr., South Carolina Electric and Gas Company, to H. R. Denton, NRC, November 4, 1983.

2.5 Trojan Nuclear Plant, 50-344, TAC No. 52890

The licensee for the Trojan Plant (Portland General Electric Co.) provided a response to Item 2.1 (Part 1) in a submittal dated November 4, 1983. The submittal states that the components whose function is required to trip the reactor are identified as safety-related on documents, procedures and in information handling systems used in the plant to control safety-related activities including maintenance, work orders and parts replacement.

2.6 Conclusion

Based on the review of the licensee's submittal, we find that the licensee's response confirms that the components required to trip the reactor are identified as safety-related, and that documents used to initiate design, maintenance, or procurement require identification of safety-related components. The licensee's response, therefore, meets the requirements of Item 2.1 (Part 1) of Generic Letter 83-28, and is acceptable.

Reference

1. Letter, B. D. Withers, Portland General Electric Co., to D. G. Eisenhut, NRC, November 4, 1983.

2.7 Yankee Rowe, 50-29, TAC No. 52895

The licensee for the Yankee Rowe Nuclear Plant (Yankee Atomic Electric Co.) provided a response to Item 2.1 (Part 1) in a submittal dated November 5, 1983. The submittal states that the reactor trip system and components whose function is required to trip the reactor are identified as safety-related on appropriate documents, procedures and information handling systems used in the plant to control safety-related activities, including maintenance, job orders, and parts replacement.

2.8 Conclusion

Based on the licensee's submittal, we find that the response meets the requirements of item 2.1 (Part 1) of Generic Letter 83-28 and is, therefore, acceptable.

Reference

1. Letter, L. H. Heider, Yankee Atomic Electric Company, to D. G. Eisenhut, NRC, November 5, 1983.

3. GENERIC REFERENCES

1. Generic Implications of ATWS Events at the Salem Nuclear Power Plant, NUREG-1000, Volume 1, April 1983; Volume 2, July 1983.

2. NRC Letter, D. G. Eisenhut to all Licensees of Operating Reactors, Applicants for Operating License, and Holders of Construction Permits, "Required Actions Based on Generic Implications of Salem ATWS Events (Generic Letter 83-28)," July 8, 1983.

APPENDIX O

EG&G TECHNICAL EVALUATION REPORT FOR SOUTH TEXAS UNITS 1 AND 2: CONFORMANCE TO GENERIC LETTER 83-28 ITEMS 3.1.3 AND 3.2.3

EGG-NTA-7159

CONFORMANCE TO GENERIC LETTER 83-28
ITEMS 3.1.3 AND 3.2.3
SOUTH TEXAS, UNIT NOS. 1 AND 2

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ABSTRACT

This EG&G Idaho, Inc., report provides a review of the submittals from South Texas, Unit Nos. 1 and 2, for conformance to Generic Letter 83-28, items 3.1.3 and 3.2.3.

Docket Nos. 50-498 and 50-499

FOREWORD

This report is supplied as part of the program for evaluating licensee/applicant conformance to Generic Letter 83-28 "Required Actions Based on Generic Implications of Salem ATWS Events." This work is being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of PWR Licensing-A by EG&G Idaho, Inc., NRR and I&E Support Branch.

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Docket Nos. 50-498 and 50-499

CONTENTS

CONFORMANCE TO GENERIC LETTER 83-28
ITEMS 3.1.3 AND 3.2.3
SOUTH TEXAS, UNIT NOS. 1 AND 2

1. INTRODUCTION

On July 8, 1983, Generic Letter No. 83-28[1] was issued by D. G. Eisenhut, Director of the Division of Licensing, Nuclear Reactor Regulation, to all licensees of operating reactors, applicants for operating licenses, and holders of construction permits. This letter included required actions based on the generic implications of the Salem ATWS events. These requirements have been published in Volume 2 of NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant".[2]

This report documents the EG&G Idaho, Inc., review of the submittals from South Texas, Unit Nos. 1 and 2, for conformance to items 3.1.3 and 3.2.3 of Generic Letter 83-28. The submittals and other documents utilized in this evaluation are referenced in Section 4 of this report.

2. REVIEW REQUIREMENTS

Item 3.1.3 (Post-Maintenance Testing of Reactor Trip System Components) requires licensees and applicants to identify, if applicable, any post-maintenance test requirements for the reactor trip system (RTS) in existing technical specifications that can be demonstrated to degrade rather than enhance safety. Item 3.2.3 applies this same requirement to all other safety-related components. Any proposed technical specification changes resulting from this action shall receive a pre-implementation review by the NRC.

The relevant submittals for South Texas, Unit Nos. 1 and 2, were reviewed to determine compliance with items 3.1.3 and 3.2.3 of the generic letter. First, the submittals from this plant were reviewed to determine that these two items were specifically addressed. Second, the submittals were checked to determine if any post-maintenance test items specified in the technical specifications were identified that were suspected to degrade rather than enhance safety. Last, the submittal was reviewed for evidence of special conditions or other significant information relating to the two items of concern.

3. REVIEW RESULTS FOR SOUTH TEXAS, UNIT NOS. 1 AND 2

3.1 Evaluation

Houston Lighting and Power Co., the applicant for South Texas, Unit Nos. 1 and 2, provided an initial response to items 3.1.3 and 3.2.3 of Generic Letter 83-28 in a submittal dated June 28, 1985.[3] In the submittal, the applicant stated that the technical specifications were being developed and that the NRC would be notified if any post-maintenance test requirements are perceived to degrade rather than enhance safety. In a subsequent submittal dated January 28, 1986,[4] the applicant stated that they had reviewed the technical specifications and that no post-maintenance test requirements were identified that degrade safety.

3.2 Conclusions

The applicant's responses to items 3.1.3 and 3.2.3 meet the requirements of Generic Letter 83-28 and are acceptable.

4. REFERENCES

1. NRC Letter, D. G. Eisenhut to all Licensees of Operating Reactors, Applicants for Operating License, and Holders of Construction Permits, "Required Actions Based on Generic Implications of Salem ATWS Events (Generic Letter 83-28)", July 8, 1983.

2. Generic Implications of ATWS Events at the Salem Nuclear Power Plant, NUREG-1000, Volume 1, April 1983; Volume 2, July 1983.

3. Houston Lighting and Power Co. letter, J. H. Goldberg to H. L. Thompson, NRC, June 28, 1985.

4. Houston Lighting and Power Co. letter, M. R. Wisenburg to V. S. Noonan, NRC, January 28, 1986.