NUREG-1137
Supplement No. 6

Safety Evaluation Report
related to the operation of Vogtle Electric Generating Plant, Units 1 and 2

Docket Nos. 50-424 and 50-425

Georgia Power Company, et al.

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

March 1987

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ABSTRACT

In June 1985, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-1137) regarding the application of Georigia Power Company, Municipal Electric Authority of Georgia, Oglethorpe Power Corporation, and the City of Dalton, Georgia, for licenses to operate the Vogtle Electric Generating Plant, Units 1 and 2 (Docket Nos. 50-424 and 50-425). Supplement 1 to NUREG-1137 was issued by the staff in October 1985, Supplement 2 was issued in May 1986, Supplement 3 was issued in August 1986, Supplement 4 was issued in December 1986, and Supplement 5 was issued in January 1987. The facility is locted in Burke County, Georgia, approximately 26 miles south-southeast of Augusta, Georgia, and on the Savannah River.

This sixth supplement to NUREG-1137 provides recent information regarding resolution of some of the open and confirmatory items that remained unresolved at the time the Safety Evaluation Report was issued.

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

APPENDICES

TABLES

1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

In June 1985, the Nuclear Regulatory Commission staff (NRC or staff) issued a Safety Evaluation Report (SER), NUREG-1137, on the application of the Georgia Power Company (hereinafter referred to as the applicant) for licenses to operate the Vogtle Electric Generating Plant, Units 1 and 2. Supplement 1 to NUREG-1137 was issued in October 1985, Supplement 2 was issued in May 1986, Supplement 3 was issued in August 1986, Supplement 4 was issued in December 1986, and Supplement 5 was issued in January 1987. This document, the sixth supplement to that SER (SSER 6), provides the staff evaluation of open and confirmatory items that have been resolved since SSER 5 was issued. This SSER is the final supplement to the SER in support of the issuance of a full-power license for Vogtle Unit 1 and provides the staff's conclusion, based on review of the Final Safety Analysis Report (FSAR) through Amendment 32, that Vogtle Unit 1 may be issued a license authorizing power up to 100%.

Each of the sections and appendices of this supplement is designated the same as the related portion of the SER. Each section is supplementary to and not in lieu of the discussion in the SER unless otherwise noted. Appendix A is a continuation of the chronology of this safety review, and Appendix B lists reference materials cited in this document.* Appendix D lists acronyms and initialisms used in this supplement, Appendix E lists the principal contributors, and Appendix K is a continuation of errata to the SER supplements. Appendices C, F, G, H, I, J, L, M, N, O, P, Q, R, S, and T have not been changed by this supplement. Appendices U and V are new as a result of this supplement. Appendix U discusses the pump and valve inservice testing program. Appendix T evaluates the safety parameter display system.

In addition to updating the status of unresolved items as identified in SSER 5, this supplement

- discusses the applicant's recent submittal on the loose parts detection program (see Section 4.4.7)
- discusses recent information provided by the applicant on nuclear service cooling water system welding (see Section 6.6.6)
- discusses resolution of the low-power license condition regarding leak rate measurements to comply with NUREG-0737, Item III.D.1.1 (see Section 11.5.3)

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

The NRC Project Manager for the operating license review is Melanie A. Miller. She may be reached by telephone at (301) 492-7000 or by mail at the following address:

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Office of Nuclear Reactor Regulation
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Copies of this SER supplement are available for inspection at the NRC Public Document Room, 1717 H Street NW., Washington, DC 20555, and at the local Public Document Room at the Burke County Library, 4th Street, Waynesboro, Georgia 30830.

1.7 Open Items

The SER identified 14 items in the staff review that had not been resolved with the applicant at the time that report was issued. SSER 1 fully resolved open items 3, 9, 10, and 12 and partially resolved open item 2. SSER 2 fully resolved open items 4 and 6; partially resolved open items 5, 7a, and 13; expanded open item 7a; and identified five new open items (7c and 15 through 18). SSER 3 fully resolved open items 7b and 18; partially resolved open items 1, 5, 7a, 11, and 13; expanded open item 1; and identified four new open items (7d and 19 through 21). SSER 4 fully resolved open items 7, 8, 11, 13, 15, 17, and 19; partially resolved open items 5 and 16; resolved open item 2 for Unit 1; partially resolved open item 1 for Unit 1; changed open item 20 to a license condition; and identified three new open items. SSER 5 fully resolved open items 1, 21, 23, and 24; partially resolved open items 5, 14a, 14b, and 22; changed the remaining portion of open item 14a to a license condition; and changed the remaining portion of open item 16 to a schedular exemption. This supplement partially resolves open item 14b and changes the remaining portion to a license condition.

The complete list of open items is reproduced in updated Table 1.4; the current status of each item is given. For those items addressed in this supplement, the relevant section is noted.

1.8 Confirmatory Items

The SER identified 50 items that required confirmatory information and hence were not fully resolved at the time that report was issued. SSER 1 fully resolved confirmatory items 6, 8, 11, 20, 32, 45, 46, and 47. SSER 2 fully resolved confirmatory items 2, 3, 5, 16, 17, 21, 31, 34, 37, 38, 44, and 50 and added confirmatory item 51. SSER 3 fully resolved confirmatory items 4, 33, 39, 41, and 51; resolved confirmatory items 18 and 23 for Unit 1 only; added items 52 and 53; and partially resolved and expanded item 14. SSER 3 also changed confirmatory item 22 to open item 19 and confirmatory items 43, 48, and 49 to license conditions. SSER 4 fully resolved confirmatory items 1, 7, 9, 10, 12, 15, 35, 42, and 53; partially resolved confirmatory item 14; partially resolved confirmatory item 36 for Unit 1; resolved confirmatory items 24, 26, and 52 for Unit 1 and 27 for Unit 2; changed confirmatory item 19 to a license condition; and opened confirmatory items 54 (Unit 2 only) and 55. SSER 5 fully resolved confirmatory items 14 and 29; resolved confirmatory items 25, 28, and 36 for Unit 1; and reopened confirmatory item 27 for

Unit 2. This supplement fully resolves confirmatory item 55 and resolves confirmatory items 13 and 30 for Unit 1.

The complete list of confirmatory items and their status is provided in updated Table 1.5. If the confirmatory item is discussed in this supplement, the section in which it is discussed is identified.

1.9 License Condition Items

In the SER, the staff identified 11 license conditions. These issues will be cited in the operating license or Technical Specifications to ensure that NRC requirements are met during plant operation unless these conditions have subsequently been resolved. SSER 1 added license condition 12. SSER 2 added license condition 13 on fire protection. SSER 3 added license conditions 14 and 15 based on previous confirmatory items (items 43, 48, and 49) and deleted license conditions 4 and 11 because the applicant had fulfilled those requirements by including them in the Technical Specifications. SSER 4 resolved license conditions 1, 5 (Unit 1), and 6 because the applicant had fulfilled the requirements of the license conditions. SSER 4 also added license conditions 16 through 19. SSER 5 resolved license conditions 2 (Unit 1), 3, 9, and 16, added license condition 20, and added as a license item a schedular exemption for spent fuel pool racks. This supplement resolves license condition 14 and adds license condition 22. Table 1.6 is updated in this report to reflect these changes.

Table 1.4 Listing of open items (revised from SSER 5)

Item	Status	Section*
(1) Equipment qualification		
(a) Seismic equipment qualification	Partially resolved and expanded (SSER 3), resolved for Unit 1 (SSER 4)	
(b) Environmental equipment qualification	Resolved for Unit 1 (SSER 5)	
(c) Pump and valve operability assurance	Partially resolved (SSER 3), resolved for Unit 1 (SSER 4)	
(i) Aging and sequence of environmental conditions in maintenance program	Resolved (SSER 4)	
(ii) Pumps affected by static shaft analysis	Resolved (SSER 3)	
(iii) Onsite audit	Resolved (SSER 4)	
(iv) Safety injection pump operation	Opened (SSER 3), resolved (SSER 4)	
(v) Demonstrate operability of check valves	Opened (SSER 3), resolved (SSER 4)	
(vi) Uniform thread engagement	Opened (SSER 3), resolved (SSER 4)	
(2) Preservice inspection program	Partially resolved (SSER 1), resolved for Unit 1 (SSER 4)	
(3) Containment sump	Resolved (SSER 1)	
(4) Toxic gas evaluation of chemicals	Resolved (SSER 2)	
(5) Generic Letter 83-28	Partially resolved (SSERs 2, 3, 4, and 5), awaiting information and under staff review	
(6) Emergency response capability--RG 1.97, Rev. 2	Resolved (SSER 2)	

*Section of this supplement in which item is discussed.

Table 1.4 (Continued)

Item	Status	Section*
(7) Fire protection items		
(a) Fire doors and dampers	Partially resolved (SSERs 2 and 3), resolved (SSER 4)	
(b) Power supplies for ventilation	Resolved (SSER 3)	
(c) Sprinkler system flushing deviation	Opened (SSER 2), resolved (SSER 4)	
(d) Fire hazards analysis	Opened (SSER 3), resolved (SSER 4)	
(8) Safe and alternate shutdown capability	Resolved (SSER 4)	
(9) Training of emergency diesel generator personnel	Resolved (SSER 1)	
(10) Diesel fuel oil storage tank cathodic protection	Resolved (SSER 1)	
(11) Licensee qualifications for operation	Partially resolved (SSER 3), resolved (SSER 4)	
(12) Retesting of simulator response (NUREG-0737, Item I.A.2.1)	Resolved (SSER 1)	
(13) Emergency preparedness	Partially resolved (SSERs 2 and 3), resolved (SSER 4)	
(14) Human factors engineering items		
(a) Detailed control room design review	Partially resolved and changed to license condition (SSER 5)	
(b) Safety parameter display system	Partially resolved (SSERs 5 and 6), changed to license condition (SSER 6)	18.2
(15) Arbitrary intermediate pipe break criteria	Opened (SSER 2), resolved (SSER 4)	

*Section of this supplement in which item is discussed.

Table 1.4 (Continued)

Item	Status	Section*
(16) Spent fuel pool rack design	Opened (SSER 2), partially resolved (SSER 4), schedular exemption (SSER 5)	
(17) Training program	Opened (SSER 2), resolved (SSER 4)	
(a) Mitigating core damage (NUREG-0737, Item I.A.2.1)		
(b) Instructor qualification and requalification		
(c) Licensed operator training		
(d) Nonlicensed personnel training		
(e) Records of plant personnel training		
(18) Compliance with RG 1.94	Opened (SSER 2), resolved (SSER 3)	
(19) LOCA mitigation in Modes 3 and 4	Opened (SSER 3), resolved (SSER 4)	
(20) Alternate radwaste facility	Opened (SSER 3), changed to license condition (SSER 4)	
(21) Physical security	Opened (SSER 3), resolved (SSER 5)	
(22) Seismic adequacy of plastic tie wraps	Opened (SSER 4), partially resolved (SSER 5)	
(23) Verification of computer codes used to analyze ASME components	Opened (SSER 4), resolved (SSER 5)	
(24) Multiple response spectrum methodology	Opened (SSER 4), resolved (SSER 5)	

*Section of this supplement in which item is discussed.

Table 1.5 Listing of confirmatory items (revised from SSER 5)

Item	Status	Section*
(1) Correlation and analysis of data from old and new meteorological towers	Resolved (SSER 4)	
(2) Upgrade of operational meteorological measurements program	Resolved (SSER 2)	
(3) Atmospheric dispersion model for dose assessments	Resolved (SSER 2)	
(4) NSCW cooling tower seepage analysis	Resolved (SSER 3)	
(5) Details of groundwater monitoring program	Resolved (SSER 2)	
(6) Verification of FSAR commitments on compaction of Category 1 backfill	Resolved (SSER 1)	
(a) Audit of compaction control records		
(b) Submittal and evaluation of supplemental test results		
(7) Submittal and evaluation of settlement records and settlement monitoring program	Resolved (SSER 4)	
(8) Foundation competency of clay marl stratum	Resolved (SSER 1)	
(9) Steamline break analysis outside of containment	Resolved (SSER 4)	
(10) Final pipewhip and jet impingement evaluation for high-energy piping	Resolved (SSER 4)	
(11) Design documents review	Resolved (SSER 1)	
(12) Compliance with NUREG-0737, Item II.D.1	Resolved (SSER 4)	
(13) Program submittal for inservice testing of pumps and valves	Resolved for Unit 1 (SSER 6)	3.9.6

*Section of this supplement in which item is discussed.

Table 1.5 (Continued)

Item	Status	Section*
(14) Pump and valve operability assurance	Partially resolved (SSERs 3 and 4), resolved (SSER 5)	
(a) Compliance with RG 1.148	Resolved (SSER 4)	
(b) Methods and standards for qualification	Resolved (SSER 3)	
(c) Qualification of pump and motor	Resolved (SSER 4)	
(d) Generic testing criteria for qualifying check valves	Resolved (SSER 4)	
(e) Administrative control of component qualification	Resolved (SSER 4)	
(f) Dependability of containment isolation (purge valves)	Resolved (SSER 5)	
(g) Long-term operability of deep draft pumps (IE Bulletin 79-15)	Resolved (SSER 4)	
(h) Issues regarding AFW turbine	Opened (SSER 3), resolved (SSER 4)	
(i) Operability of the feedwater check valve	Opened (SSER 3), resolved (SSER 4)	
(j) FSAR list of active pumps and valves	Opened (SSER 3), resolved (SSER 4)	
(k) Preservice tests prior to fuel load	Opened (SSER 3), resolved (SSER 4)	
(l) Complete qualification prior to fuel load	Opened (SSER 3), resolved (SSER 4)	
(15) Compliance with NUREG-0737, Item II.F.2	Resolved (SSER 4)	
(16) Conformance to 10 CFR 50, Appendix G, criteria (PORV setpoint curve)	Resolved (SSER 2)	
(17) Discrepancy between WCAP-10529 and FSAR	Resolved (SSER 2)	
(18) Examination of steam generator tubes	Resolved for Unit 1 (SSER 3)	

*Section of this supplement in which item is discussed.

Table 1.5 (Continued)

Item	Status	Section*
(19) Natural circulation boration and cooldown tests	Changed to license condition (SSER 4)	
(20) Target Rock valves in RVHVS	Deleted as errata (SSER 1)	
(21) Containment responses following an MSLB	Resolved (SSER 2)	
(22) Operator action in event of a small-break LOCA	Changed to open item 19 (SSER 3)	
(23) Volumetric examination of engineered safety features systems	Resolved for Unit 1 (SSER 3)	
(24) Test of engineered safeguards P-4 interlock	Resolved for Unit 1 (SSER 4)	
(25) IE Bulletin 80-06 concerns	Resolved for Unit 1 (SSER 5)	
(26) Override of isolation signals	Resolved for Unit 1 (SSER 4)	
(27) Bypass and inoperable status panel	Resolved (SSER 4), reopened (SSER 5)	
(28) Compliance with NUREG-0737, Item II.K.3.1	Resolved for Unit 1 (SSER 5)	
(29) Capacity of each reserve auxiliary transformer to start and run the loads of both Class 1E trains	Resolved (SSER 5)	
(30) Verification test results for the adequacy of plant electric distribution system voltages	Resolved for Unit 1 (SSER 6)	8.4.1
(31) Coordination and testing of circuit breakers located in the primary circuit of regulated transformers used as isolation devices	Resolved (SSER 2)	

*Section of this supplement in which item is discussed.

Table 1.5 (Continued)

Item	Status	Section*
(32) Plant-specific procedure for estimating core damage (NUREG-0737, Item II.B.3)	Resolved (SSER 1)	
(33) Demonstrate effective communications	Resolved (SSER 3)	
(34) Offsite communications	Resolved (SSER 2)	
(35) Procedures for load following test and/or maintenance	Resolved (SSER 4)	
(36) Incorporation of generic and plant-specific recommendations for TDI diesel generators	Partially resolved for Unit 1 (SSER 4), resolved for Unit 1 (SSER 5)	
(37) Procedures for ordering fuel after 5 days	Resolved (SSER 2)	
(38) Procedures for general housekeeping and maintenance	Resolved (SSER 2)	
(39) Process control program	Resolved (SSER 3)	
(40) Volume reduction system (VRS)	Under staff review (see SSER 5 Section 11.4)	
(a) VRS topical report		
(b) Potential accidents involving the VRS		
(c) VRS inputs		
(d) VRS filter testing		
(41) Compliance with NUREG-0737, Item II.K.3.31	Resolved (SSER 3)	
(42) Compliance with NUREG-0737, Item II.F.1	Resolved (SSER 4)	
(43) Compliance with NUREG-0737, Item III.D.1.1	Changed to license condition (SSER 3)	
(44) Procedures generation package	Resolved (SSER 2)	

*Section of this supplement in which item is discussed.

Table 1.5 (Continued)

Item	Status	Section*
(45) Program to minimize post-LOCA leakage from ESF system outside containment	Deleted as errata (SSER 1)	
(46) Analysis of dropped control rod event for DNB limits	Resolved (SSER 1)	
(47) Inadvertent boron dilution during modes 3, 4, and 5	Resolved (SSER 1)	
(48) Operator action in event of an SGTR	Changed to license condition (SSER 3)	
(49) Radiological consequences of an SGTR	Changed to license condition (SSER 3)	
(50) Program to minimize ECCS equipment leakage	Resolved (SSER 2)	
(51) Generic Letter 85-12	Opened (SSER 2), resolved (SSER 3)	
(52) Seismic equipment qualification	Opened (SSER 3), resolved for Unit 1 (SSER 4)	
(a) Stress criteria for emergency and faulted conditions		
(b) Completion of seismic qualification program		
(c) Verification of as-built loads		
(d) FSAR revisions		
(53) Emergency preparedness	Opened (SSER 3), resolved (SSER 4)	
(54) Implementation of seismic separation program for Unit 2	Opened (SSER 4), awaiting information	
(55) Annunciator for high-flow signal to isolate electric steam boiler line	Opened (SSER 4), resolved (SSER 6)	7.6.2.3

*Section of this supplement in which item is discussed.

Table 1.6 Listing of license conditions (revised from SSER 5)

Item	Status	Section*
(1) Long-term groundwater and settlement monitoring requirements	Resolved (SSER 4)	
(2) Inservice testing of pumps and valves	Resolved for Unit 1 (SSER 5)	
(3) Final baseline report for the loose parts monitoring system	Resolved (SSER 5)	
(4) Technical Specification for maximum permissible temperature mismatch	Resolved (SSER 3)	
(5) Inservice inspection program	Resolved for Unit 1 (SSER 4)	
(6) Operability requirements for vent system in Technical Specifications	Resolved (SSER 4)	
(7) Exemption from 10 CFR 50, Appendix J, Paragraph III.D.2(b)(ii)		
(8) Exemption from 10 CFR 70.24		
(9) Operating experience on shift	Resolved (SSER 5)	
(10) Implementation and maintenance of physical security plan		
(11) Technical Specification to require four valves to be closed during refueling	Resolved (SSER 3)	
(12) Reactor vessel level instrumentation system implementation report		
(13) Fire protection		
(14) Receipt of leak rate test results	Resolved (SSER 6)	11.5.3
(15) Steam generator tube rupture		
(16) Natural circulation boration and cooldown tests	Resolved (SSER 5)	
(17) Replacement of zinc coating of Unit 1 diesel fuel oil storage tanks		
(18) TDI maintenance and surveillance items		

*Section of this supplement in which item is discussed.

Table 1.6 (Continued)

Item	Status	Section*
(19) Monitoring of alternate radwaste facility exhaust		
(20) Detailed control room design review		
(21) Schedular exemption for spent fuel pool racks		
(22) Safety parameter display system	Added (SSER 6)	18.2

*Section of this supplement in which item is discussed.

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.9 Mechanical Systems and Components

3.9.6 Inservice Testing of Pumps and Valves

In SSER 5, the staff stated that the applicant had submitted a program for the inservice testing (IST) of pumps and valves for Unit 1 by letter dated July 30, 1986, and that this submittal was reviewed by the staff and was the subject of a working meeting with the applicant on October 8 and 9, 1986. On the basis of staff comments during the meeting, the applicant submitted a revised IST program by letter dated October 31, 1986. This revision superseded the previous submittal and included changes resulting from discussions during the October 8 and 9, 1986 meeting.

As stated in SSER 5, the applicant's IST program was submitted in accordance with the requirements of 10 CFR 50.55a(g) to comply with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI. The IST program was prepared for Vogtle Unit 1 in accordance with the requirements of Sections IWP and IWV of Section XI, 1983 Edition through the Summer 1983 Addenda. Pursuant to 10 CFR 50.55a(g)(5), the applicant has requested relief from certain ASME Code testing requirements for specific pumps and valves when the Code requirements are impractical within the limits of design, geometry, and system safety. The applicant's request for relief includes an explanation and justification for the relief and a proposal for alternative test procedures.

The staff and its contractor, EG&G Idaho, Inc., have completed the review of the Vogtle Unit 1 IST program and find the program and associated requests for relief from certain Code requirements acceptable with two exceptions. The detailed review of the Unit 1 program is presented in the EG&G Technical Evaluation Report (TER) dated February 1987, which is included as Appendix U to this report. The staff will review the Unit 2 IST program upon its submittal. The exceptions are summarized below, and the applicant is required to resolve these exceptions as stated.

(1) The applicant requested relief from measuring flow rate for boric acid transfer pumps on the basis that flow instrumentation is not being installed and that pump degradation would be detected by measuring changes in differential pressure of a fixed resistance flow path. The staff finds that the fixed-resistance-flow-path method gives an indication of pump operability. However, limiting measurements to changes in differential pressure only may not be sufficient to detect the pump degradation because changes in differential pressure may result from changes of flow-path resistances without indicating whether or not the pump is degrading. Because differential pressure changes due to degradation of the fixed-resistance flow path are not expected to occur until after a significant period of operation, the staff finds this method acceptable until the first refueling outage. The applicant is, therefore, required to install instruments to measure the Code-required flow rate before restart

following the first refueling outage. (See Section 2.2.1 of the TER in Appendix U.)

(2) The applicant requested relief from the increased test frequency requirement for certain degraded valves and proposed to test degraded valves at each cold shutdown rather than the Code-required monthly frequency. The intent of the Code requirement is to increase the test frequency of degraded valves so that immediate action can be taken if needed. The applicant's proposal would leave degraded valves untested during power operation between cold shutdowns. Because the interval between cold shutdowns is likely to be much longer than 1 month, the risk of a totally failed valve, when called on to function, might increase to an unacceptable level. The staff, therefore, concludes that the applicant's proposal is unacceptable and that degraded valves must be tested each month or repaired and tested before returning to power if the affected valves can only be tested and repaired during cold shutdowns. (See Section 3.1.3 of the TER in Appendix U.)

4 REACTOR

4.4 Thermal-Hydraulic Design

4.4.7 Loose Parts Monitoring System

In the SER, the staff indicated that the applicant needed to provide, before power operation, a final baseline report containing the following: (1) an evaluation of the Vogtle loose parts monitoring system (digital metal impact monitoring system (DMIMS)) for conformance to Regulatory Guide (RG) 1.133, "Loose Part Detection Program for the Primary System of Light-Water-Cooled Reactors"; (2) a description of the system hardware, operation, and implementation of the loose parts detection program, including plans for startup testing, acquisition of baseline data, and alarm settings; and (3) a description and evaluation of diagnostic procedures used to confirm the presence of a loose part.

By letter dated February 24, 1987, the applicant provided the above information with the exception of actual Vogtle baseline data. The applicant stated that these data must be collected at the 100% power level and would be provided to the staff within 90 days after reaching 100% power.

The applicant referred to FSAR Section 1.9 for a discussion of conformance of the DMIMS to RG 1.133. The FSAR section indicates conformance except for a lack of description of sensor locations in the plant Technical Specifications. However, this information is not part of the standard Technical Specifications and is therefore acceptable.

The system hardware is discussed in FSAR Section 4.4.6.4, question 492.1, and a letter dated October 12, 1984. The system design was found acceptable by the staff as discussed in the SER. System operation is discussed in four procedures addressing operation and operator response, channel calibration, and daily surveillance. A preoperational test was performed on the DMIMS to establish initial alarm setpoints and initial calibration. Final calibration was determined through the initial test program.

The data collection will follow the appropriate Westinghouse procedure, and the data will be generated from a series of simulated impacts made at each location at a determined distance from each accelerometer.

The diagnostic process upon suspicion of a loose part includes engineering evaluation with Westinghouse assistance. Such an evaluation would consider comparison with baseline data. Required corrective action would consider object mass, magnitude of impact, location of impact site, mobility, repetition rate, and retrieval expense vs. potential damage.

Based on a review of the applicant's February 24, 1987, submittal and its commitment to provide the actual baseline data within 90 days of achieving 100% power, the staff concludes that the DMIMS and its implementation and planned operation form an acceptable loose parts detection program.

6 ENGINEERED SAFETY FEATURES

6.6 Inservice Inspection of Class 2 and 3 Components

6.6.6 Compliance With 10 CFR 50.55a(e)

According to 10 CFR 50.55a(e)(2), Class 3 components are required to be fabricated, designed, and inspected to the rules of Paragraph NCA-1140 of Section III of the ASME Boiler and Pressure Vessel Code. This paragraph of the ASME Code allows the owner to designate the Code edition and addenda to be included in the design specification. The applicant has designated that the Class 3 piping in the Vogtle nuclear service cooling water (NSCW) system be designed, fabricated, and inspected to the Winter 1977 Addenda to the 1977 Edition of the ASME Code. Alternative requirements to the ASME Code are permitted in accordance with the criteria in 10 CFR 50.55a(a)(3). This paragraph of 10 CFR 50 permits proposed alternatives to the requirements in 10 CFR 50.55a(e) when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant must demonstrate that the proposed alternative would provide an acceptable level of quality and safety or that compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Through its Quality Concern Program, the applicant received a concern (Quality Concern No. 86V0515) regarding an unconsumed insert on a Unit 2 NSCW weld. In response to Quality Concern No. 86V0515, the applicant initiated a reinspection of all accessible Class 3 stainless steel Schedule 10 piping over 12 in. in diameter in the Unit 2 NSCW system. The reinspection also included six 8-in. welds and one 3-in. weld. The reinspection reports indicate that 15 welds had lack of fusion (penetration) between the insert and pipe. This condition results in a crevice being formed between the unconsumed insert and the pipe. The discrepancy reports and the applicant's analysis of the flaws are contained in a letter from R. E. Conway to Dr. J. Nelson Grace dated January 13, 1987. An additional fracture mechanics analysis and an inservice examination program were submitted for staff review by letter dated February 9, 1987.

The ASME Code requires that the flaws observed during the reinspection be eliminated, reduced to an acceptable limit, or repaired when necessary. In lieu of meeting these requirements, the applicant performed a fracture mechanics analysis to demonstrate that the flaws will not affect the serviceability of the components and proposed an inservice examination program.

The fracture mechanics evaluation contains a fatigue growth analysis, a limit load analysis, and an evaluation of the flaws to the acceptance criteria of Paragraph IWB-3640 of the 1986 Edition of ASME Code Section XI. The fatigue evaluation was performed using the methodology in Appendix A to ASME Code Section XI. The fatigue evaluation indicates that flaw growth resulting from fatigue would be insignificant. The fatigue evaluation used the "reference fatigue crack growth curves" for carbon and low-alloy ferritic steels. Because the welds in the NSCW system were fabricated using SA 312 Type 304L austenitic material and not carbon and low-alloy ferritic steels, the applicant's evaluation has not used the correct crack growth curves. However, based on its experience with fatigue evaluations of austenitic material and nuclear industry experience with service water systems, the staff believes that the flaws in the austenitic NSCW piping will not have significant growth from fatigue.

The limit load analysis is based on the methodology in Appendix C to Section XI of the ASME Code. The limit load analysis indicates that a through-wall crack approximately 70 degrees around the pipe circumference and a full-circumference part-through crack with a depth equal to 43% of the wall thickness would meet ASME Code limits for the service conditions specified for the piping. Using the maximum stress calculated in accordance with ASME Code Section III, Paragraph ND-3652, and the method of analysis documented in Paragraph IWB-3640 of ASME Code Section XI, the allowable full-circumference flaw size is at least 41% of the wall thickness. The flaw size estimated from inspection and sample testing was 31% of the pipe wall thickness. Because the estimated flaw size is less than the acceptable flaw size calculated using ASME Code criteria, the fracture mechanics evaluation indicates that the piping is acceptable for service.

The applicant has also evaluated the susceptibility of the Vogtle NSCW system to intergranular stress corrosion cracking (IGSCC) and crevice corrosion. The applicant indicates that IGSCC has not been a problem with low carbon grade stainless steels such as Type 304L, which is used in the Vogtle NSCW system. Based on industry experience, the applicant states that IGSCC and crevice corrosion will not occur for the design and operating conditions of the Vogtle NCSW system.

The applicant concludes that the initial flaw will not grow during the life of the plant, and the NSCW piping is acceptable for the service life of Vogtle. The staff agrees that the observed flaws will not immediately affect the serviceability of the NSCW piping. However, the staff is uncertain about the amount of growth these flaws will experience during service. The fracture mechanics evaluation indicates that if the flaws were to either grow to full circumference with a depth of 41% of the wall or to through the wall and 70 degrees around the circumference, the serviceability of piping could be affected. Hence, flaw growth could affect the serviceability of the NSCW lines. Because lack of fusion results in a crevice between the insert and the pipe, these types of flaws could be a source of accelerated flaw growth. In addition, accelerated flaw growth has been observed in service water systems at Salem Unit 2 and Peach Bottom Unit 2. Based on its experience with other nuclear service water systems and the nature of the flaws in the Vogtle service water system, the staff believes that the rate of flaw growth should be monitored at Vogtle. In its February 9, 1987, letter, the applicant proposed an inservice examination program consisting of pipe leak and ultrasonic examination during the first 10 years of service. Specifically, the applicant will perform a "walkdown" of the accessible ASME Code Class 3 portions of the NSCW piping system at operational pressures at each refueling outage for the first 10 years of service. Additionally, the applicant will perform an ultrasonic examination of one representative weld in each unit (the worst weld in Unit 2 and the corresponding Unit 1 weld) every 40 months for the first 10 years of service. This inservice examination program will monitor flaw growth and ensure pipe integrity.

Based on the fracture mechanics evaluation and an NSCW inservice examination program to monitor flaw growth, it is not necessary at this time to remove the flaws in the Vogtle NSCW piping. By monitoring flaw growth and performing the fracture mechanics evaluation, the applicant has demonstrated compliance with the criteria 10 CFR 50.55(a)(3), and the Vogtle NSCW lines are acceptable for service. The staff will evaluate further actions beyond the 10-year period should flaw growth propagate at an unacceptable rate or to an unacceptable level.

7 INSTRUMENTATION AND CONTROLS

7.6 Interlock Systems Important to Safety

7.6.2 Specific Findings

7.6.2.3 Instrumentation for Process Measurements Used for Safety Functions

In SSER 4, the staff indicated that the applicant had committed to install an annunciator for the high-flow signal which isolates the electric steam boiler line before full-power operation of Unit 1 and before fuel load of Unit 2. Installation of the annunciator was identified as a confirmatory item. The applicant, by letters dated February 18 and 26, 1987, stated that the annunciator has been installed on the common annunciator panel in the Unit 1 portion of the control room. Because the purpose of the common annunciator panel is to contain the annunciators for shared systems and the electric steam boiler system is a shared system, the applicant stated that a separate annunciator would not be installed in the Unit 2 portion of the control room. The staff has reviewed the information on the installation and concludes that it fully resolves confirmatory item 55 for both units.

8 ELECTRIC POWER SYSTEMS

8.4 Other Electrical Features and Requirements for Safety

8.4.1 Adequacy of Station Electric Distribution System Voltages

In the SER, the staff evaluated the Vogtle design for conformance to the positions in Branch Technical Position (BTP) PSB-1 (NUREG-0800). The staff identified a confirmatory item regarding the results of a verification test performed by the applicant to substantiate the accuracy of the Vogtle voltage analysis in conformance with Position 4 of BTP PSB-1. As stated in SSER 5, the applicant had indicated that test results would be provided before 5% power is exceeded.

By letter dated March 9, 1987, the applicant provided the results of this test, which was performed in accordance with the guidance of Position 4 of BTP PSB-1. The results indicate that the measured voltages are no more than 1.0% (steady state) and 2.7% (transient) below the analytically derived voltages. These values are less than the maximum value of 3% allowed by Position 4 of BTP PSB-1. Also, when the test results are used to modify the original voltage analyses, there is no detrimental impact on the voltages supplied to the loads. The staff therefore considers the verification test results to be satisfactory. This resolves confirmatory item 30 for Unit 1. Before licensing of Unit 2, the applicant must perform the same verification test of that unit.

11 RADIOACTIVE WASTE MANAGEMENT

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

11.5.3 TMI-2 Action Plan Requirements

Item II.D.1.1 Integrity of Systems Outside Containment Likely to Contain Radioactive Material

In SSER 3, the staff reported that the program to control leakage was consistent with NRC criteria and was acceptable but that the required leak rate measurement data had to be submitted. The applicant committed to submit the required data before 5% of full power was exceeded, and the submittal was made a condition of the low-power license.

By letter dated March 9, 1987, the applicant submitted the necessary leak rate measurement data. The following data were reported:

(1) residual heat removal (RHR) system - 0.00202 gpm

(2) containment spray system (excluding NaOH subsystem) - 0.000326 gpm

(3) safety injection system (excluding boron injection and accumulators) - 0 gpm.

(4) chemical and volume control system (CVCS) (letdown, boron recycle, and charging pumps) - 0.00557 gpm.

(5) postaccident sampling system

- liquid leakage - 0.001 gpm
- gaseous leakage - 22 standard cubic centimeter (no leakage identified external to the system)

(6) gaseous waste processing system

- liquid leakage - 0.001 gpm
- gaseous leakage - 0 standard cubic centimeter (external to the system)

(7) nuclear sampling system (pressurizer steam and liquid sample lines, reactor coolant sample lines, RHR sample lines, and CVCS demineralizer and letdown heat exchanger sample lines only)

- liquid leakage - 0.001 gpm
- gaseous leakage (steam) - 0 gpm

Data were not submitted for the positive displacement pump (PDP) because this pump could not be placed in service. The applicant committed to submit the PDP leakage measurements by April 15, 1987. On the basis of discussions of the PDP problem with the resident inspector, the staff concludes that delay in submitting these data is justified. The staff will review the PDP data when they become available.

The staff has reviewed the leak rate measurement data submitted and concludes that the low values of the data show that the program to control leakage is effective. The staff concludes that this satisfies the requirement of the applicable low-power license condition.

18 HUMAN FACTORS ENGINEERING

18.2 Safety Parameter Display System

Each operating reactor shall be provided with a safety parameter display system (SPDS) in the control room. The Commission requirements for the SPDS are defined in NUREG-0737, Supplement 1.

The purpose of the SPDS is to provide a concise display of critical plant variables to control room operators to aid them in rapidly and reliably determining the safety status of the plant. NUREG-0737, Supplement 1 requires licensees and applicants to prepare a written safety analysis describing the basis on which the selected parameters are sufficient to assess the safety status of each identified function for a wide range of events, which includes symptoms of severe accidents. Licensees and applicants shall also prepare an implementation plan for the SPDS which contains schedules for design, development, installation, and full operation of the SPDS as well as a design verification and validation (V&V) plan. The safety analysis and the implementation plan are to be submitted to the staff for review and discussion in an SER.

The staff review for licensees requesting a preimplementation review and for all applicants consists of a review of SPDS documentation (i.e., safety analysis report and implementation plan) and audit meetings and site visits.

After an initial review of the utility's submittals, three separate audit meetings and site visits, as described below, may be arranged. As dictated by the comprehensiveness of the utility's documentation and the schedule for design and implementation of the SPDS, the objectives of these audits may be met in fewer site visits.

The purpose of the design verification audit meeting is to obtain additional information required to (1) resolve any outstanding questions about the V&V program, (2) confirm that the V&V program is being correctly implemented, and (3) audit the results of the V&V activities to date. At this meeting, the utility should provide a thorough description of the SPDS design process. Emphasis should be placed on how the utility is ensuring that the implemented SPDS will provide appropriate parameters, be isolated from safety systems, provide reliable and valid data, and incorporate good human engineering practices.

After review of all documentation, a design validation audit may be conducted to review the as-built prototype or installed SPDS. The purpose of this audit is to ensure that the results of the utility's testing demonstrate that the SPDS meets the functional requirements of the design and to ensure that the SPDS exhibits good human engineering practices.

As necessary, a final installation audit may be conducted at the site to ascertain if the SPDS has been installed in accordance with the utility's plan and is functioning properly. A specific purpose is to ensure that the data displayed reflect the sensor signal which measures the variable displayed.

This audit will be coordinated with and may be conducted by the NRC Resident Inspector.

Unlike licensees, applicants will undergo, before implementation, a full review to determine whether the applicable provisions of NUREG-0737, Supplement 1 have been satisfied. To the extent possible, the staff will temper its review to conform to the schedule for licensing and SPDS implementation.

Because the Vogtle SPDS was in an advanced stage of development when the staff's review began, a combined design verification and design validation audit was conducted on December 3-4, 1986.

By letters dated September 27, 1985, May 29, October 31, and December 22, 1986, and January 5 and 12, 1987, the applicant provided documentation regarding the SPDS for Vogtle.

A staff concern identified during its review was that because the V&V of the plant safety monitoring system (PSMS) software, which provides input to the SPDS, was not completed, the potential exists that the SPDS might present invalid data that could mislead the operator. By letter dated January 12, 1987, the applicant committed to perform a weekly check whereby selected control board-displayed data would be compared with similar PSMS data to demonstrate acceptability of PSMS-displayed values. In addition, the applicant committed to develop acceptance criteria that instruct the operator as to when the PSMS-displayed information is unacceptable and hence when to declare the SPDS inoperable whenever the PSMS-displayed information is questionable. This issue is discussed in detail in Section 7.5.2.1 of SSER 5.

The Vogtle SPDS is part of the Vogtle emergency response facility computer system (ERFCS). The ERFCS receives analog inputs from several plant systems, including the PSMS and the plant effluent radiation monitoring system (PERMS).

The control room SPDS workstation contains two displays. There are three levels of displays for the critical safety functions (CSFs):

(1) Top-level displays show the value of specific top-level parameters in the form of color-coded deviation bar charts or tabular displays and status of engineered safeguards and CSF status.

(2) Second-level displays depict a logic path for each critical safety function status tree (CSFST) (a CSFST corresponds to each CSF except radiation monitoring).

(3) Third-level displays include time-history and parameter vs. parameter plots, numeric indication of CSF parameter values, and status indication for SPDS discrete inputs.

Color coding of continuously displayed CSF status boxes and logic trees is used to indicate the status of each CSF. Visual alarms (status box flashing and appropriate color coding of status boxes, logic trees, etc.) are presented to alert operators that alarm threshold values have been exceeded.

User access to CSF-related displays is via a keypad configured to correspond from top to bottom to the hierarchy of CSF importance and, except for radiation monitoring, in a left-to-right fashion corresponding to the display hierarchy. Below the keys that control access to CSF-related keys is a grouping of keys that provides access to radiation monitoring second- and third-level displays. A single key press will access any SPDS display.

SPDS displays can be called up on any of three SPDS terminals in the control room, five SPDS terminals in the Technical Support Center, and four terminals in the Emergency Operations Facility.

The staff conclusions with regard to each element of the SPDS required by NUREG-0737, Supplement 1 are summarized below and discussed in detail in the Technical Evaluation Report (TER) reproduced in Appendix V of this report. Where there are differences between the SER and the TER, the SER is the prevailing document.

(1) <u>The SPDS should provide a concise display</u>.

The Vogtle SPDS provides concise displays thus meeting the requirements of NUREG-0737, Supplement 1.

(2) <u>The SPDS should display critical plant variables (parameter selection)</u>.

Based on applicant submittals and the results of the onsite audit, the staff concludes that the CSF parameters selected and displayed at Vogtle are appropriate and will provide the operators with the status of the CSFs thus meeting the requirements of NUREG-0737, Supplement 1.

(3) <u>The SPDS is to have rapid and reliable display of the safety status of the plant</u>.

The Vogtle SPDS has for the most part satisfied the requirements of NUREG-0737, Supplement 1 regarding rapid and reliable display of SPDS information. However, for the staff to complete its review the applicant should provide a report addressing the following by March 1, 1988. Submittal of this report is a license condition.

- Establish and implement realistic rather than arbitrary criteria for interchannel comparison of redundant inputs. These values must be appropriate both for adverse and normal operating conditions, and must be based on anticipated instrument loop accuracies. (Refer to Section 4.3 of Appendix V.)

- Provide and discuss system availability.

(4) <u>The principal purpose of the SPDS is to aid control room personnel during abnormal and emergency conditions</u>.

The Vogtle SPDS adequately aids the control room personnel in determining the safety status of the plant and meets the requirements of NUREG-0737, Supplement 1.

(5) <u>The SPDS shall be located convenient to the control room operators</u>.

The Vogtle SPDS visual display terminals are located in the control room, do not interfere with operator movement, and meet the requirements of NUREG-0737, Supplement 1.

(6) <u>The SPDS shall continuously display information from which the safety status of the plant can be assessed</u>.

The Vogtle SPDS has continuous displays and meets the requirements of NUREG-0737, Supplement 1.

(7) <u>The SPDS shall be suitably isolated from electrical or electronic interference</u>.

The isolation of the SPDS at Vogtle meets the requirements of NUREG-0737, Supplement 1. Section 18.2 of SSER 5 discusses SPDS isolation in detail.

(8) <u>Procedures should be developed and operators should be trained to respond to accident conditions both with and without the SPDS available</u>.

The Vogtle SPDS meets this NUREG-0737, Supplement 1 requirement.

(9) <u>The SPDS display shall be designed to incorporate human factors principles</u>.

The Vogtle SPDS generally incorporates accepted human factors engineering principles; however, the consultant's TER in Appendix V identifies a number of areas in which design improvements would be likely to enhance SPDS usability. Because of the large number of items identified in the TER, the staff has concluded that in these areas human factors engineering principles should be more fully addressed. Each of the items discussed below should be evaluated and its final disposition discussed in a report to the staff by March 1, 1988. Submittal of this report is a license condition.

- In the SPDS color-coding scheme, perceptual cues for challenges to CSFs are lost when a CSF parameter is of questionable validity. Some other cue should be provided to indicate questionable data.
- The containment isolation valve status display uses the color codes red for open and green for closed. This is consistent with the convention for valve position lights, but is not consistent with the SPDS convention of green for safe, red for unsafe. Conversely, use of the green/safe, red/unsafe convention would violate the valve status color convention. Operator input should be used in determining which convention is adopted.
- Parameter alarm status is shown as green for normal, red for high, and flashing red for high-high or low-low. Parameter alarm color coding might be more easily understandable if the CSF color-coding scheme of green-normal, yellow-alert, orange-severe challenge, and red-unsafe is used.

- Few prompts are currently presented. Required user responses might be less ambiguous if prompts were used to guide parameter value selection with keyboard arrow keys and to guide numerical inputs via keyboard.

- The color of indicated setpoints and data plots is sometimes the same, making discrimination difficult or impossible.

- Acceptable operating levels are often not indicated on graphic displays.

- Default values are generally not presented.

- Sometimes the underline cursor which is displayed is difficult to locate. The use of a block cursor should be considered as a solution.

- Displays may contain numerous numerical values, some of which may be selected to bring up additional data screens, and some of which may not. Differential coding of selectable and nonselectable values would avoid erroneous selections.

- Indication of current parameter values should be presented on status tree displays.

- The cursor often moves to a location from which it must be moved for data input or selection of options. Unnecessary, additional interaction steps could be eliminated if a cursor could move directly to an active data input or option selection area.

- Scroll keys would be easier to use if the forward and backward scroll keys were appropriately labeled.

- User errors and uncertainty about the results of a selection might be reduced if parameter values selected by users (to produce subsequent screens) were displayed in reverse video for a second or two immediately after users designate such a selection through cursor positioning.

In addition to the above nine requirements, Section 18.2 of the Standard Review Plan (NUREG-0800) specifies that the staff will review the applicant's SPDS V&V programs. Accordingly, the following steps were taken by the applicant as discussed in its October 31, 1986, submittal:

(1) A system requirement was established by the applicant.

(2) A design verification review was conducted for the applicant by Energy, Inc. Deficiencies identified by this review were either corrected by the applicant or justification for not correcting the deficiency was documented. However, this review did not include the PERMS and the PSMS which support the ERFCS. To complete the design verification activity of the SPDS, the applicant committed to complete this program by June 1, 1987, by letter dated December 22, 1986. Completion of this program should include operator feedback as discussed in Section 3.3.2 of the TER in Appendix V of this supplement as well as V&V of the PERMS.

(3) The applicant conducted integrated systems validation testing which confirmed that the system performance meets the functional system specifications. However, this testing did not validate the CSFST logic nor did it prompt validation test participants to identify SPDS improvements. Therefore, the validation does not constitute a complete and rigorous man-in-the-loop (operator participation and feedback in the entire process) test of the SPDS. The applicant should develop a process to obtain operator feedback and to solicit prompt user comments and opinions.

In summary, the applicant has installed an SPDS which meets the requirements of NUREG-0737, Supplement 1, except where noted above. The staff did not identify any serious safety concern; thus, the SPDS may be considered fully operational. However, certain information noted above must be provided to satisfy the applicable license condition and to allow the staff to complete its review of the Vogtle SPDS. Before licensing of Unit 2, the applicant must demonstrate similarity of the Unit 1 and Unit 2 systems. The staff will perform further review of the Unit 2 SPDS as necessary.

APPENDIX A

CONTINUATION OF CHRONOLOGY OF RADIOLOGICAL REVIEW OF VOGTLE UNITS 1 AND 2 OPERATING LICENSE REVIEW

December 5, 1986	Letter from applicant forwarding FSAR Amendment 30.
December 23, 1986	Letter from applicant concerning deferral of pre-operational testing.
December 30, 1986	Letter from applicant concerning Technical Specifications.
December 31, 1986	Letter from applicant concerning operator licensing examination site visits requirements.
January 2, 1987	Letter from applicant concerning SER open item 14a, "Detailed control room design review."
January 2, 1987	Letter from applicant concerning Technical Specifications.
January 5, 1987	Letter from applicant concerning emergency plan implementation procedures: nuclear operations.
January 5, 1987	Letter from applicant concerning Offsite Dose Calculation Manual.
January 7, 1987	Atomic Safety and Licensing Appeal Board (ASLAB) issues Notice of Oral Argument on the appeal of Georgians Against Nuclear Energy from the Atomic Safety Licensing Board's (ASLB's) August 27, 1986, partial initial decision.
January 7, 1987	Letter from applicant concerning completion of cable vendor survey for ethylene vinyl acetate insulation.
January 15, 1987	Letter from applicant concerning SER open item 5, "Generic Letter 83-28."
January 16, 1987	ASLAB issues Order stating that there is no bar to issuance of an operating license by the Director of Nuclear Reactor Regulation.
January 16, 1987	Letter from applicant forwarding Revision 9 to the Emergency Plan.
January 16, 1987	Letter to applicant forwarding Facility Operating License NPF-61 authorizing operation at up to 5% of full power.

January 19, 1987	Letter from applicant concerning emergency plan implementation procedures: nuclear operations.
January 21, 1987	Letter from applicant concerning SER open item 16, "Spent fuel pool rack design."
January 21, 1987	ASLAB issues Memorandum and Order explaining and affirming January 16, 1987, Order.
January 22, 1987	Letter from applicant concerning Technical Specifications.
February 2, 1987	ASLAB issues Order stating that it will review sua sponte the ASLB December 23, 1986, Concluding Partial Initial Decision.
February 4, 1987	Letter from applicant concerning piping penetration area filtration and exhaust system Technical Specification change.
February 9, 1987	Letter from applicant concerning nuclear service cooling water systems.
February 12, 1987	Letter from applicant concerning emergency plan implementing procedure transmittal.
February 16, 1987	Letter from applicant concerning outstanding submittals list.
February 18, 1987	Letter from applicant concerning confirmatory item 55, "Annunciator for high-flow signal to isolate electric steam boiler line."
February 24, 1987	Letter from applicant concerning digital metal impact monitoring system.
February 26, 1987	Letter from applicant concerning confirmatory item 55, "Annunciator for high-flow signal to isolate electric steam boiler line."
February 27, 1987	Letter from applicant forwarding proposed revisions to full-power Technical Specifications.
March 6, 1987	Letter from applicant forwarding proposed revisions to full-power Technical Specifications.
March 9, 1987	Letter from applicant concerning leak rate measurements associated with license condition on NUREG-0737, Item III.D.1.1.
March 9, 1987	Letter from applicant concerning SER confirmatory item 30, "Verification test results for the adequacy of plant electric distribution system voltages."

APPENDIX B

REFERENCES

Code of Federal Regulations, Title 10, "Energy," U.S. Government Printing Office, Washington DC, revised annually.

Conway, R. E., GPC, letter to J. Nelson Grace, NRC, "Nuclear Service Cooling Water Welding Quality Concern," January 13, 1987.

U.S. Nuclear Regulatory Commission, Bulletin 79-15, "Deep Draft Pump Deficiencies," July 11, 1979.

---, Bulletin 80-06, "Engineered Safety Feature (ESF) Reset Controls," March 13, 1980.

---, Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," July 8, 1983.

---, Generic Letter 85-12, "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps'," June 28, 1985.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980; Supplement 1, January 1983.

---, NUREG-0800 (formerly NUREG-75/087), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," July 1981.

Westinghouse Electric Corp., WCAP-10529, R. Fleming, "COMS, Cold Overpressure Mitigating Systems," February 1984.

INDUSTRY CODE

American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Personnel," Paragraph NCA-1140.

---, Section III, Paragraph ND-3652.

---, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix A.

---, Section XI, Appendix C.

---, Section XI, Paragraph IWB-3640, 1986 Edition.

---, Section XI, Sections IWP and IWX, 1983 Edition through Summer 1983 Addenda.

---, Winter 1977 Addenda to 1977 Edition.

APPENDIX D

ACRONYMS AND INITIALISMS

AFW	auxiliary feedwater
ASLAB	Atomic Safety and Licensing Appeal Board
ASLB	Atomic Safety and Licensing Board
ASME	American Society of Mechanical Engineers
BTP	branch technical position
CFR	Code of Federal Regulations
CSF	critical safety function
CSFST	critical safety function status tree
CVCS	chemical and volume control system
DMIMS	digital metal impact monitoring system
DNB	departure from nucleate boiling
ECCS	emergency core cooling system
ERFCS	emergency response facility computer system
ESF	engineered safety feature
FSAR	Final Safety Analysis Report
IE	Office of Inspection and Enforcement
IGSCC	intergranular stress corrosion cracking
IST	inservice testing
LOCA	loss-of-coolant accident
MSLB	main steamline break
NRC	U.S. Nuclear Regulatory Commission
NSCW	nuclear service cooling water
PERMS	plant effluent radiation monitoring system
PORV	power-operated relief valve
PSMS	plant safety monitoring system
RG	regulatory guide
RHR	residual heat removal
RVHVS	reactor vessel head vent system
SER	Safety Evaluation Report
SGTR	steam generator tube rupture
SPDS	safety parameter display system
SSER	Supplement to Safety Evaluation Report

ACRONYMS AND INITIALISMS (Continued)

TDI	Transamerica Delaval, Inc.
TER	Technical Evaluation Report
TMI-2	Three Mile Island, Unit 2
VRS	volume reduction system
V&V	verification and validation

APPENDIX E

NRC STAFF CONTRIBUTORS AND CONSULTANTS

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

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APPENDIX K

ERRATA TO SUPPLEMENTS 4 and 5 TO THE VOGTLE SAFETY EVALUATION REPORT

Page	Paragraph*	Line	Change
Supplement 4			
1-2	8	11	Change "three" to "four".
1-3	-	7	Delete "14,".
1-3	-	8	Add "partially resolves confirmatory item 14;" after "53;"
1-3	2	8	Add "Unit 1" after "5".
1-6	-	Last	Add "(25) Replacement of zinc coating of Unit 2 diesel fuel oil storage tanks Added (SSER 4) 9.5.4.2".
1-12	-	Item 17	Change "9.5.4.1" to "9.5.4.2".
Supplement 5			
1-2	4	12	Change to "items 1, 21, 23, and 24; partially resolves open items 5, 14a, 14b, and 22; changes the remaining portion of open item 14a to a license condition; and changes the remaining portion of open item 16 to a schedular exemption."
1-3	3	9	Delete "14,".
1-3	3	17	Change "56" to "55".
1-3	5	11	Add ", adds license condition 20, and adds as a license item a schedular exemption for spent fuel pool racks". after "and 16".
1-5	-	26	Change "Resolved" to "Partially resolved and changed to license condition".

*Full paragraph

Page	Paragraph*	Line	Change
Supplement 5 (Continued)			
1-9	-	Item (28)	Add "for Unit 1" after "Resolved".
1-12	-	-	Insert after Item (19) "(20) Detailed control room design review Added (SSER 5) 18.1".
1-12	-	-	Insert after Item (20) "(21) Schedular exemption for spent fuel pool racks Added (SSER 5) 9.1.2".

*Full paragraph

APPENDIX U

TECHNICAL EVALUATION REPORT OF THE PUMP AND VALVE INSERVICE TESTING PROGRAM VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

EGG-NTA-7468

TECHNICAL EVALUATION REPORT
PUMP AND VALVE INSERVICE TESTING PROGRAM
VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

Docket No. 50-424

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ABSTRACT

This EG&G Idaho, Inc., report presents the results of our evaluation of the Vogtle Electric Generating Plant, Unit 1, Inservice Testing Program for safety-related pumps and valves.

FOREWORD

This report is supplied as part of the "Review of Pump and Valve Inservice Testing Programs for Operating License Plants" Program 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 S[illegible]

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Docket No. 50-424

CONTENTS

TECHNICAL EVALUATION REPORT

PUMP AND VALVE INSERVICE TESTING PROGRAM

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

1. INTRODUCTION

Contained herein is a technical evaluation of the pump and valve inservice testing (IST) program submitted by the Georgia Power Company for its Vogtle Electric Generating Plant, Unit 1.

By a letter dated July 30, 1986 Georgia Power Company submitted an IST program for Vogtle Electric Generating Plant, Unit 1. The working session with Georgia Power Company and Southern Company Services representatives was conducted on October 8 and 9, 1986. The applicant's revised program, as attached to J. A. Bailey letter to NRC, dated October 31, 1986, which supercedes the previous submittal, was reviewed to verify compliance of proposed tests of Class 1, 2, and 3 safety related pumps and valves with the requirements of the ASME Boiler and Pressure Vessel Code (the Code), Section XI, 1983 Edition through Summer 1983 Addenda. Any IST program revisions subsequent to those noted above are not addressed in this technical evaluation report (TER). It is an NRC staff position that required program changes, such as additional relief requests or the deletion of any components from the IST program, should be submitted to the NRC under separate cover in order to receive prompt attention, but should not be implemented prior to review and approval by the NRC.

In their submittal Georgia Power Company has requested relief from the ASME Code testing requirements for specific pumps and valves and these requests have been evaluated individually to determine whether they are indeed impractical. This review was performed utilizing the acceptance criteria of the Standard Review Plan, Section 3.9.6, and the Draft Regulatory Guide and Value/Impact Statement titled "Identification of Valves for Inclusion in Inservice Testing Programs". These IST program testing requirements apply only to component testing (i.e., pumps and valves) and are not intended to provide the basis to change the applicant's current Technical Specifications for system test requirements.

Section 2 of this report presents the Georgia Power Company bases for requesting relief from the Section XI requirements for the Vogtle Electric Generating Plant, Unit 1 pump testing program and EG&G's evaluations and conclusions regarding these requests. Similar information is presented in Section 3 for the valve testing program.

The NRC staff's positions and guidelines concerning inservice testing requirements are provided in Appendix A.

Category A, B, and C valves that meet the requirements of the ASME Code, Section XI, and are not exercised quarterly are discussed in Appendix B.

A listing of P&IDs used for this review is contained in Appendix C.

Inconsistencies and omissions in the applicant's program noted in the course of this review are listed in Appendix D. The applicant should resolve these items in accordance with the evaluations, conclusions, and guidelines presented in this report.

2. PUMP TESTING PROGRAM

The Vogtle Electric Generating Plant, Unit 1 IST program submitted by the Georgia Power Company was examined to verify that all pumps that are included in the program are subjected to the periodic tests required by the ASME Code, Section XI, 1983 Edition through Summer of 1983 Addenda, except for those pumps identified below for which specific relief from testing has been requested. Each Georgia Power Company basis for requesting relief from the pump testing requirements and the EG&G reviewer's evaluation of that request is summarized below.

2.1 All Systems (except Nuclear Service Cooling Water System)

2.1.1 Relief Request

The applicant has requested relief from the Table IWP-3100-1 requirement of Section XI for measurement of pump bearing temperature yearly.

2.1.1.1 Applicant's Basis for Requesting Relief. The yearly temperature measurement will not provide significant information about pump conditions. Industry experience has shown that bearing temperature changes caused by degrading bearings occur only after major degradation has occurred at the pump. Prior to this major pump degradation, the vibration measurement would provide the necessary information to warn of an impending malfunction. Deletion of this measurement will not have a significant effect on pump evaluation since vibration amplitude is measured quarterly.

2.1.1.2 Evaluation. The reviewer agrees with the applicant that quarterly measurement of pump bearing vibration displacement will provide earlier indication of bearing problems that may result from pump bearing degradation than the annual measurement of bearing temperatures. The reviewer also agrees that changes in bearing or lubricant temperatures as a result of bearing problems usually occur only after significant degradation of the pump bearing has already occurred. The quarterly bearing vibration monitoring would have detected the degradation long before the increase temperatures were noticed thus deletion of the annual bearing temperature measurement will have no significant effect on pump performance evaluation.

2.1.1.3 Conclusion. The reviewer concludes that the vibration amplitude measurement will provide the necessary information to warn of an impending pump malfunction hence the deletion of yearly pump bearing temperature measurement will not have a significant effect on pump performance evaluation. The reviewer concludes that the other required testing will give reasonable assurance of pump operability required by the Code and, therefore, relief should be granted.

2.2 Chemical and Volume Control System

2.2.1 Relief Request

The applicant has requested relief from the IWP-3100 requirement of Section XI for the boric acid transfer pumps for measurement of pump flow rate.

The applicant has requested relief from the IWP-3100 requirement of Section XI for the boric acid transfer pumps for varying system resistance to obtain the reference value of either measured differential pressure or measured flow rate and proposed to utilize a closed-loop fixed-resistence recirculation flow path to determine pump degradation.

2.2.1.1 Applicant's Basis for Requesting Relief. Relief is requested from measuring pump flow rate as the plant does not have permanent flow rate measuring instruments.

Relief is requested from varying the resistance of the system as the test flow path, utilizing flow orifice FO-10117 to and from the boric acid storage tank, is a fixed resistance test flow path and not a variable resistance test flow path. During preoperational testing the flow rate from pumps 1-1208-P6-006 and 1-1208-P6-007 was measured to be 30.5 gpm and

31.5 gpm, respectively. This established the reference value flow rates for these pumps with their corresponding differential pressure measurements. During inservice testing pump degradation would be detected by changes in differential pressure and flow rate measurements would be unnecessary.

2.2.1.2 Evaluation. The reviewer does not agree with the applicant that not having permanent flow rate measurement instruments negates the requirement to measure pump flow rate. The NRC staff position is that lack of instrumentation is not sufficient justification to not measure Code required parameters.

The reviewer agrees with the applicant that utilizing the fixed resistance test flow path to achieve conditions for measurement of Code required parameters will provide sufficient information to monitor for pump degradation.

With the addition of the installed flowrate instrumentation mentioned above, the measurement of both flowrate and differential pressure and the use of a band of acceptance criteria for variations in the two measured parameters should provide for detection of degradation of these pumps.

2.2.1.3 Conclusion. The reviewer concludes that not having permanent flow rate measuring instruments does not negate the requirement to measure pump flow rate, therefore, relief from measuring boric acid transfer pumps flow rate should not be granted.

The reviewer concludes that utilization of a fixed resistance test flow path to achieve conditions for measurement of Code required parameters is adequate, therefore, relief from utilization of a variable resistance flow path should be granted.

2.2.2 Relief Request

The applicant has requested relief from the IWP-4120 requirement of Section XI for the boric acid transfer pumps suction pressure gauges to have a full scale range of three times the reference value or less.

2.2.2.1 Applicant's Basis for Requesting Relief. Suction pressure gauges PI-10115 and PI-10116 on the boric acid transfer pumps have a range of 0 psi to 15 psi. The suction pressure measurements taken during preoperational testing were between 2 nd 3 psi. Therefore, the maximum full scale range of the gauge would h e to be from 0 to 6 or 9 psi to be within Code requirements. These instruments are within the accuracies of Table IWP-4110-1. Considering the low pressure involved, the difference between the Code ranges and the range on the installed instruments would have no significance on the adequacy of the measurements taken. The installed instruments will be used for taking suction pressure measurements during pump tests.

2.2.2.2 Evaluation. The reviewer agrees with the applicant that the installed boric acid transfer pumps suction pressure gauges are sufficient to measure Code required pump suction pressure and that the variance on the range of the installed instruments would have no significant effect on the adequacy of the measurement.

2.2.2.3 Conclusion. The reviewer concludes that utilization of the installed boric acid transfer pumps suction pressure gauges is adequate to measure Code required pump inlet pressure, therefore, relief should be granted.

3. VALVE TESTING PROGRAM

The Vogtle Electric Generating Plant, Unit 1 IST program submitted by the Georgia Power Company was examined to verify that all valves that are included in the program are subjected to the periodic tests required by the ASME Code, Section XI, 1983 Edition through Summer 1983 Addenda, and the NRC positions and guidelines. The reviewers found that, except as noted in Appendix D or where specific relief from testing has been requested, these valves are tested to the Code requirements and the NRC positions and guidelines summarized in Appendix A. Each Georgia Power Company basis for requesting relief from the valve testing requirements and the reviewer's evaluation of that request is summarized below and grouped according to system and valve category.

3.1 All Systems

3.1.1 Corrective Action

3.1.1.1 Relief Request. The applicant has requested relief from testing all valves that require corrective action as a result of cold shutdown and refueling outage testing in accordance with the requirements of Section XI, Paragraphs IWV-3417(b) and IWV-3523 and proposed to utilize plant Technical Specifications to control whether plant startup is permissible or not.

3.1.1.1.1 Applicant's Basis for Requesting Relief--The plant Technical Specifications provide the requirements and plant conditions necessary for plant startup (i.e., mode changes). As an alternative, the test requirement will be satisfied before the valve is required to be operable in accordance with the plant Technical Specifications.

3.1.1.1.2 Evaluation--The reviewer agrees with the applicant that the plant Technical Specifications dictate the necessary requirements and plant conditions for plant startup (i.e., mode changes). The plant Technical Specifications place adequate controls on system and/or valve operability by establishing and defining the Limiting Conditions for Operation which restrict, allow, or require entry into the various modes of plant operation. However, any valve that is inoperable prior to plant startup and cannot be tested prior to return to service and is subsequently required by Technical Specifications during operation shall be repaired prior to startup (see Section 3.1.3 of this report).

3.1.1.1.3 Conclusion--The reviewer concludes that the applicant's Technical Specifications dictate the necessary requirements and plant conditions for startup and operations. The Section XI requirements determine component operability status and should not preclude plant startup when all applicable Technical Specifications requirements are met. The reviewer concludes that the testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.1.2 Rapid Acting Valves

3.1.2.1 Relief Request. The applicant has requested relief from the power operated valve stroke time trending requirements of Section XI, Paragraph IWV-3417(a), for all rapid-acting, power operated valves whose function is safety related and proposed to apply a maximum stroke time limit of 2 seconds to all rapid-acting, power operated valves; i.e., those valves with normal stroke times of less than 2 seconds. This includes reactor coolant system power operated relief valves 1201-PV-0455A and 0456A.

3.1.2.1.1 Applicant's Basis for Requesting Relief--These solenoid-operated valves have very short stroke times and are classified as "rapid-acting" valves. Accurate measurement of stroke time is not practical. In addition, stroke times may vary significantly due to system pressure and/or temperature changes from one test to another. As an alternative, these valves will be required to be full-stroked and timed to the nearest second quarterly. Acceptance of the test will be based only on the stroke time limit (not to exceed 2 seconds) and not on the "50%" criteria of IWV-3417.

3.1.2.1.2 Evaluation--The reviewer agrees with the applicant's proposal to place a 2 second maximum limit on stroke time for rapid acting power operated valves. This proposal is consistent with the NRC staff position on rapid acting valves discussed in Appendix A, Section 8 of this report.

3.1.2.1.3 Conclusion--The reviewer concludes that the applicant's proposal to assign a maximum stroke time limit of 2 seconds on their rapid acting power operated valves is in accordance with the NRC staff's position on rapid acting valves and should be sufficient to determine proper valve operability. The reviewer concludes that this alternate criteria proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.1.3 Valves Tested During Cold Shutdown

3.1.3.1 Relief Request. The applicant has requested relief from the corrective action requirement of IWV-3417(a) for category A and category B valves identified as being tested on a cold shutdown frequency (Appendix B of this report) and proposed to modify the required monthly testing frequency for degraded valves to a cold shutdown frequency.

3.1.3.1.1 Applicant's Basis for Requesting Relief--Valves that are normally tested during cold shutdown cannot be tested once each month. Stroking these valves during power operation may place the plant in an unsafe condition. As an alternative, the test frequency shall be increased to once each cold shutdown, not to exceed once each month.

3.1.3.1.2 Evaluation--The reviewer does not agree with the applicant's basis for requesting relief from the increased test frequency requirements of Section XI for those valves that are specifically identified for testing only during cold shutdowns. The Code requires an increased frequency of tests to assure continued operability of the degraded valves to demonstrate valve operability. Valves that are specifically identified for testing only during cold shutdowns and refueling outages that are found to have exceeded the allowable change in stroke time and cannot be tested at the increased frequency should be repaired and demonstrated operable prior to being required for operation by the plant Technical Specifications.

3.1.3.1.3 Conclusion--The reviewer concludes that the applicant's proposal to test degraded cold shutdown exercised valves on a cold shutdown frequency will not be sufficient to demonstrate proper compliance with the corrective action requirements of IWV-3417(a). The reviewer concludes that the alternate testing proposed will not give reasonable assurance of valve operability as required by the Code and, therefore, relief should not be granted.

3.2 Reactor Coolant System

3.2.1 Category A/C Valves

3.2.1.1 Relief Request. The applicant has requested relief from exercising valve U6 112, reactor makeup water to pressurizer relief tank check, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by full-stroke exercising this valve on a refueling outage frequency.

3.2.1.1.1 Applicant's Basis for Requesting Relief--This check valve cannot be exercised during power operation or cold shutdown as the only method available to verify reverse flow closure is valve leak testing during Appendix J, Type C, leak testing during refueling outages. As an alternative reverse flow closure will be verified during Appendix J, Type C, leak testing during refueling outages.

3.2.1.1.2 Evaluation--The reviewer agrees with the applicant that valve U6 112 cannot be full- or partial-stroke exercised during power operation or cold shutdown due to the fact that the only method available to verify reverse flow closure is valve leak testing during Appendix J, Type C, leak testing during refueling outages.

3.2.1.1.3 Conclusion--The reviewer concludes that the applicant's proposal to verify closure capability of valve U6 112 on a refueling outage frequency should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.3 Safety Injection System

3.3.1 Category C Valves

3.3.1.1 Relief Request. The applicant has requested relief from exercising valves U4 026, 027, 028, 029, and U6 013, boron injection to cold leg checks, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by full-stroke exercising these valves on a refueling outage frequency.

3.3.1.1.1 Applicant's Basis for Requesting Relief--These check valves cannot be exercised during power operation as the only method available to verify full flow operability is by using charging pump flow through the boron injection tank into the cold legs. This, however, exposes the safety injection nozzles to thermal shock and unnecessarily changes reactor coolant system boron concentration. These check valves cannot be exercised during cold shutdown as charging pump flow could result in a low temperature overpressurization of the reactor coolant system (RCS). As an alternative these check valves will be full-stroke exercised during refueling outages when the reactor vessel head is removed and full charging pump flow can be utilized.

3.3.1.1.2 Evaluation--The reviewer agrees with the applicant that valves U4 026, 027, 028, 029, and U6 013 cannot be full-stroke exercised during power operation due to safety injection nozzle thermal shock considerations and unnecessary RCS boron concentration changes. The reviewer agrees with the applicant that these valves cannot be full-stroke exercised during cold shutdown due to possible low temperature overpressurization of the RCS.

3.3.1.1.3 Conclusion--The reviewer concludes that the applicant's proposal to full-stroke exercise valves U4 026, 027, 028, 029, and U6 013 on a refueling outage frequency should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.3.1.2 Relief Request. The applicant has requested relief from exercising valve U6 090, safety injection system (SIS) pump suction from the refueling water storage tank (RWST) check and valves U6 098 and 099, SIS pumps discharge checks, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by partial-stroke exercising these valves quarterly and full-stroke exercising these valves on a refueling outage frequency.

3.3.1.2.1 Applicant's Basis for Requesting Relief--These check valves cannot be exercised during power operation as the SIS pumps cannot overcome RCS operating pressure. These check valves cannot be exercised during cold shutdown as SIS pump flow could result in a low temperature overpressurization of the RCS. As an alternative these check valves will be partial-stroke exercised quarterly and full-stroke exercised during refueling outages when the reactor vessel head is removed and full SIS pump flow can be utilized.

3.3.1.2.2 Evaluation--The reviewer agrees with the applicant that valves U6 090, 098, and 099 cannot be full-stroke exercised during power operation due to the fact that the SIS pumps do not have the capability to full flow into the RCS when the RCS is at normal operating pressure. The reviewer agrees with the applicant that these valves cannot be full-stroke exercised during cold shutdown due to possible RCS low temperature overpressurization.

3.3.1.2.3 Conclusion--The reviewer concludes that the applicant's proposal to partial-stroke exercise quarterly and full-stroke exercise valves U6 090, 098, and 099 on a refueling outage frequency should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.3.1.3 Relief Request. The applicant has requested relief from exercising valve U6 163, residual heat removal (RHR) to SIS pump suction check, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by full-stroke exercising this valve on a refueling outage frequency.

3.3.1.3.1 Applicant's Basis for Requesting Relief--This check valve cannot be exercised during power operation as the SIS pumps cannot overcome RCS operating pressure. This check valve cannot be exercised during cold shutdown as SIS pump flow could result in a low temperature overpressurization of the RCS. As an alternative this check valve will be full-stroke exercised during refueling outages when the reactor vessel head is removed and full SIS pump flow can be utilized.

3.3.1.3.2 Evaluation--The reviewer agrees with the applicant that valve U6 163 cannot be full-stroke exercised during power operation due to the fact that the SIS pumps do not have the capacity to full flow into the RCS when the RCS is at normal operating pressure. The reviewer agrees with the applicant that these valves cannot be full-stroke exercised during cold shutdown due to possible RCS low temperature overpressurization.

3.3.1.3.3 Conclusion--The reviewer concludes that the applicant's proposal to full-stroke exercise valve U6 163 on a refueling outage frequency should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.3.1.4 <u>Relief Request</u>. The applicant has requested relief from exercising valves U4 262 and 263, sludge mixing isolation to RWST checks, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by sample disassembly/inspection on a refueling outage frequency.

3.3.1.4.1 <u>Applicant's Basis for Requesting Relief</u>--Reverse flow closure of these check valves can be verified only by disassembly and observation of the disk position. As an alternative one of these valves will be disassembled and manually stroked during refueling outages on a staggered test basis. If disassembly reveals that the valve is inoperable, the other valve will be disassembled.

3.3.1.4.2 <u>Evaluation</u>--The reviewer agrees with the applicant that valves U4 262 and 263 can only be verified to close by valve disassembly.

The NRC staff has concluded that a valve sampling disassembly/inspection utilizing a manual full-stroke of the disk is an acceptable method to verify a check valve's full-stroke capability. The sampling technique requires that each valve in the group must be of the same design (manufacturer, size, model number and materials of construction) and must have the same service conditions. Additionally, at each disassembly it must be verified that the disassembled valve is capable of full-stroking and that its internals are structurally sound (no loose or corroded parts).

A different valve of each group is required to be disassembled, inspected and manually full-stroked at each refueling, until the entire group has been tested. If it is found that the disassembled valve's full-stroke capability is in question, the remainder of the valves in that group must also be disassembled, inspected, and manually full-stroked at the same outage.

3.3.1.4.3 Conclusion--The reviewer concludes that the applicant's proposal to perform sample disassembly/inspection on a refueling outage frequency, when performed in accordance with the previous discussion (Section 3.3.1.4.2), should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability required by the Code and, therefore, relief should be granted.

3.3.2 Category A/C Valves

3.3.2.1 Relief Request. The applicant has requested relief from exercising valves U4 120, 121, 122, 123, U6 124, and 127, SIS hot leg checks and valves U4 143, 144, 145, and 146, SIS cold leg checks, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by full-stroke exercising these valves on a refueling outage frequency.

3.3.2.1.1 Applicant's Basis for Requesting Relief--These check valves cannot be exercised during power operation as the SIS pumps cannot overcome RCS operating pressure. These check valves cannot be exercised during cold shutdown as SIS pump flow could result in a low temperature overpressurization of the RCS. As an alternative these check valve will be full-stroke exercised during refueling outages when the reactor vessel head is removed and full SIS pump flow can be utilized. The total flow from one safety injection pump will be compared to the system flow balance requirements of the Technical Specifications to verify that these valves open to perform their function. The emergency core cooling system test line subsystem provides the capability for determination of the integrity of the high pressure boundaries. The subsystem is used to verify that each of the series check valves can independently sustain operational differential pressure and is closed.

3.3.2.1.2 Evaluation--The reviewer agrees with the applicant that valves U4 120, 121, 122, 123, 143, 144, 145, 146, U6 124, and 127 cannot be full-stroke exercised during power operation due to the fact that the SIS pumps do not have the capacity to full flow into the RCS when the

RCS is at normal operating pressure. The reviewer agrees with the applicant that these valves cannot be full-stroke exercised during cold shutdown due to possible RCS low temperature overpressurization.

3.3.2.1.3 Conclusion--The reviewer concludes that the applicant's proposal to full-stroke exercise valves U4 120, 121, 122, 123, 143, 144, 145, 146, U6 124, and 127 on a refueling outage frequency should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.3.2.2 Relief Request. The applicant has requested relief from exercising valves U6 079, 080, 081, and 082, accumulator outlet checks, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by sample disassembly/inspection on a refueling outage frequency.

3.3.2.2.1 Applicant's Basis for Requesting Relief--These check valves cannot be exercised during power operation as the 650 psig nitrogen charged accumulators cannot overcome normal RCS pressure in order to inject their contents into the RCS. These check valves cannot be exercised during cold shutdown as accumulator flow could result in a low temperature overpressurization of the RCS. As an alternative one of these valves will be disassembled and manually stroked during refueling outages on a staggered test basis. If disassembly reveals that the valve is inoperable, the remaining valves will be disassembled.

3.3.2.2.2 Evaluation--The reviewer agrees with the applicant that valves U6 079, 080, 081, and 082 cannot be full-stroke exercised during power operation due to insufficient accumulator discharge pressure. The reviewer agrees with the applicant that these valves cannot be full-stroke exercised during cold shutdown due to possible RCS low temperature overpressurization.

The NRC staff has concluded that valve disassembly/inspection using a manual full-stroke of the disk is an acceptable method to verify the full-stroke capability of check valves. At each disassembly the applicant must verify that the disassembled valve is capable of full-stroking and that its internals are structurally sound (no loose or corroded parts).

3.3.2.2.3 Conclusion--The reviewer concludes that the applicant's proposal to perform sample disassembly/inspection of valves U6 079, 080, 081, and 082 on a refueling outage frequency, when performed in accordance with the previous discussion (Section 3.3.1.4.2) should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.3.2.3 Relief Request. The applicant has requested relief from exercising valves U6 083, 084, 085, and 086, accumulator and RHR checks, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by partial-stroke exercising these valves during cold shutdown and by sample disassembly/inspection on a refueling outage frequency.

3.3.2.3.1 Applicant's Basis for Requesting Relief--These check valves cannot be exercised during power operation as the 650 psig nitrogen charged accumulators cannot overcome normal RCS pressure in order to inject their contents into the RCS. These check valves cannot be exercised during cold shutdown as accumulator flow could result in a low temperature overpressurization of the RCS. As an alternative these valves will be partial-stroke exercised during cold shutdown and one of these valves will be disassembled and manually stroked during refueling outages on a staggered test basis. If disassembly reveals that the valve is inoperable, the remaining valves will be disassembled.

3.3.2.3.2 Evaluation--The reviewer agrees with the applicant that valves U6 083, 084, 085, and 086 cannot be full-stroke exercised during power operation due to insufficient accumulator discharge pressure. The reviewer agrees with the applicant that these valves cannot be full-stroke exercised during cold shutdown due to possible RCS low temperature overpressurization.

The NRC staff has concluded that valve disassembly/inspection using a manual full-stroke of the disk is an acceptable method to verify the full-stroke capability of check valves. At each disassembly the applicant must verify that the disassembled valve is capable of full-stroking and that its internals are structurally sound (no loose or corroded parts).

3.3.2.3.3 Conclusion--The reviewer concludes that the applicant's proposal to partial-stroke exercise during cold shutdown and to perform sample disassembly/inspection of valves U6 083, 084, 085, and 086 on a refueling outage frequency, when performed in accordance with the previous discussion (Section 3.3.1.4.2) should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.4 Containment Spray System

3.4.1 Category C Valves

3.4.1.1 Relief Request. The applicant has requested relief from exercising valves U6 001 and 008, RWST to containment spray pump checks, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by partial-stroke exercising these valves quarterly and by sample disassembly/inspection on a refueling outage frequency.

3.4.1.1.1 Applicant's Basis for Requesting Relief--These check valves cannot be exercised during power operation as the test flow path precludes full flow testing due to pipe sizing. These check valve cannot be exercised during cold shutdown as the required recirculation flow path would cause extensive damage to components inside containment. As an alternative these valves will be partial-stroke exercised quarterly and one of these valves will be disassembled and manually stroked during refueling outages on a staggered test basis. If disassembly reveals that the valve is inoperable, the other valve will be disassembled.

3.4.1.1.2 Evaluation--The reviewer agrees with the applicant that valves U6 001 and 008 cannot be full-stroke exercised during power operation due to insufficiency of the test flow path capacity to allow full flow through the valves. The reviewer agrees with the applicant that these valves cannot be full-stroke exercised during cold shutdown due to the fact that the only full flow test path is into the containment spray header which will spray into containment thus damaging containment equipment.

The NRC staff has concluded that valve disassembly/inspection using a manual full-stroke of the disk is an acceptable method to verify the full-stroke capability of check valves. At each disassembly the applicant must verify that the disassembled valve is capable of full-stroking and that its internals are structurally sound (no loose or corroded parts).

3.4.1.1.3 Conclusion--The reviewer concludes that the applicant's proposal to partial-stroke exercise quarterly and to perform sample disassembly/inspection of valve U6 001 and 008 on a refueling outage frequency, when performed in accordance with the previous discussion (Section 3.3.1.4.2) should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.4.2 Category A/C Valves

3.4.2.1 Relief Request. The applicant has requested relief from exercising valves U6 015 and 016, containment spray checks, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by Appendix J, Type C, leak testing and by sample disassembly/inspection on a refueling outage frequency.

3.4.2.1.1 Applicant's Basis for Requesting Relief--These check valves cannot be exercised during power operation and cold shutdown as the only available flow test method would cause extensive damage to containment components. As an alternative these valves will be Appendix J, Type C, leak tested and one of these valves will be disassembled and manually stroked during refueling outages on a staggered test basis. If disassembly reveals that the valve is inoperable, the other valve will be disassembled.

3.4.2.1.2 Evaluation--The reviewer agrees with the applicant that valves U6 015 and 016 cannot be full-stroke exercised during power operation and cold shutdown due to containment equipment damage.

The NRC staff has concluded that valve disassembly/inspection using a manual full-stroke of the disk is an acceptable method to verify the full-stroke capability of check valves. At each disassembly the applicant must verify that the disassembled valve is capable of full-stroking and that its internals are structurally sound (no loose or corroded parts).

3.4.2.1.3 Conclusion--The reviewer concludes that the applicant's proposal to Appendix J, Type C, leak test and to perform sample disassembly/inspection of valves U6 015 and 016 on a refueling outage frequency, when performed in accordance with the previous discussion (Section 3.3.1.4.2) should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.5 Chemical and Volume Control System

3.5.1 Category B Valves

3.5.1.1 Relief Request. The applicant has requested relief from fail-safe testing valves HV 0190A and B, centrifugal charging pump to regenerative heat exchanger isolations, in accordance with the requirements of Section XI, Paragraph IWV-3415 and proposed to verify valve operability by full-stroke exercising and stroke timing these valves quarterly.

3.5.1.1.1 Applicant's Basis for Requesting Relief--The safety related position of these valves is open. To fail-safe test these valves to the closed position does not stroke the valve in the direction required to perform a safety related function. Therefore, a fail-safe test is not necessary. As an alternative these valves will be exercised and timed every quarter to ensure that they will perform their safety related function.

3.5.1.1.2 Evaluation--The reviewer agrees with the applicant that fail-safe testing valves HV 0190A and B serves no purpose as these valves fail closed when fail-safe tested and the safety related position for these valves is open. The reviewer agrees with the applicant that performing a fail-safe test on these valves is not necessary. Since these valves do not have a required fail-safe position, this relief request is not necessary and should be deleted.

3.5.1.1.3 Conclusion--The reviewer concludes that the applicant's proposal to full-stoke exercise and stroke time valves HV 0190A and B quarterly should be sufficient to demonstrate proper valve operability. The reviewer concludes that this relief request is unnecessary and should be deleted.

3.5.2 Category C Valves

3.5.2.1 Relief Request. The applicant has requested relief from exercising valves U6 142 and 149, charging pumps outlet checks, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by partial-stroke exercising these valves quarterly and by full-stroke exercising these valves on a refueling outage frequency.

3.5.2.1.1 Applicant's Basis for Requesting Relief--These check valves cannot be exercised during power operation as the normal charging flow path is only capable of partial-stroking them. Alternate charging flow paths cannot be utilized due to safety injection nozzle thermal shock prohibitions. These check valves cannot be exercised during cold shutdown as charging pump flow could result in a low temperature overpressurization of the RCS. As an alternative these valves will be partial-stroke exercised quarterly and full-stroke exercised during refueling outages when the reactor vessel head is removed and full charging pump flow can be utilized.

3.5.2.1.2 Evaluation--The reviewer agrees with the applicant that valves U6 142 and 149 cannot be full-stroke exercised during power operation due to chemical and volume control system (CVCS) alignment and thermal shock considerations. The reviewer agrees with the applicant that these valves cannot be full-stroke exercised during cold shutdown due to possible RCS low temperature overpressurization.

3.5.2.1.3 Conclusion--The reviewer concludes that the applicant's proposal to partial-stroke exercise quarterly and full-stroke exercise valves U6 142 and 149 on a refueling outage frequency should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.5.2.2 Relief Request. The applicant has requested relief from exercising valves U6 189 and 436, charging pump suction from the RWST check and charging pump suction from the RHR system check, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by partial-stroke exercising these valves during cold shutdown and by full-stroke exercising these valves on a refueling outage frequency.

3.5.2.2.1 Applicant's Basis for Requesting Relief--These check valves cannot be exercised during power operation or cold shutdown as both charging pumps would be required for proper flow, which would result in RCS overpressurization. Partial exercising by operating one charging pump is undesirable due to resultant RCS boron concentration changes, which could cause a plant shutdown. As an alternative these valves will be partial-stroke exercised during cold shutdown and full-stroke exercised during refueling outages when the reactor vessel head is removed and full charging pump flow can be utilized.

3.5.2.2.2 Evaluation--The reviewer agrees with the applicant that valves U6 189 and 436 cannot be full-stroke exercised during power operation due to RCS overpressurization and undesirable RCS boron concentration changes. The reviewer agrees with the applicant that these valves cannot be full-stroke exercised during cold shutdown due to RCS overpressurization.

3.5.2.2.3 Conclusion--The reviewer concludes that the applicant's proposal to partial-stroke exercise during cold shutdown and full-stroke exercise valves U6 189 and 436 on a refueling outage frequency should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.5.3 Category A/C Valves

3.5.3.1 Relief Request. The applicant has requested relief from exercising valve U6 032, CVCS to regenerative heat exchanger check, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by verifying valve closure on a refueling outage frequency.

3.5.3.1.1 Applicant's Basis for Requesting Relief--This check valve cannot be exercised during power operation as the only method available to verify reverse flow closure is valve leak testing during Appendix J, Type C, leak testing during refueling outages. As an alternative reverse flow closure will be verified during Appendix J, Type C, leak testing during refueling outages.

3.5.3.1.2 Evaluation--The reviewer agrees with the applicant that valve U6 032 cannot be verified to close during power operation or cold shutdown due to the fact that the only method available to verify reverse flow closure is valve leak testing during Appendix J Type C leak testing during refueling outages.

3.5.3.1.3 Conclusion--The reviewer concludes that the applicant's proposal to verify closure of valve U6 032 on a refueling outage frequency should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.6 Auxiliary Component Cooling Water System

3.6.1 Category C Valves

3.6.1.1 Relief Request. The applicant has requested relief from exercising valves U4 084, 085, 086, and 087, auxiliary component cooling water to reactor coolant pump (RCP) thermal barrier checks, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve closure by full-stroke exercising these valves on a refueling outage frequency.

3.6.1.1.1 Applicant's Basis for Requesting Relief--These check valves cannot be exercised during power operation as interruption of RCP thermal barrier cooling water could damage the thermal barriers. These check valves cannot be exercised during cold shutdown as installation and removal of test equipment could delay plant startup. As an alternative these valves will be full-stroke exercised closed during refueling outages.

3.6.1.1.2 Evaluation--The reviewer agrees with the applicant that valves U4 084, 085, 086, and 087 cannot be full- or partial-stroke exercised closed during power operation due to required RCP thermal barrier cooling water flow. The reviewer agrees with the applicant that these valves cannot be full- or partial-stroke exercised closed during cold shutdown due to plant startup considerations.

3.6.1.1.3 Conclusion--The reviewer concludes that the applicant's proposal to full-stroke exercise closed valves U4 084, 085, 086, and 087 on a refueling outage frequency should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.7 Main Steam System

3.7.1 Category C Valves

3.7.1.1 Relief Request. The applicant has requested relief from exercising valve U4 008, steam to auxiliary feedwater (AFW) pump check, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by partial-stroke exercising this valve quarterly and full-stroke exercising this valve open during cold shutdown. Reverse flow closure will be verified on a refueling outage frequency by disassembly/inspection.

3.7.1.1.1 Applicant's Basis for Requesting Relief--This check valve cannot be exercised during power operation as the required steam flow would inject cold auxiliary feedwater into the steam generator(s) which would result in thermal shock to steam generator internals. Reverse flow closure for this valve cannot be verified by flow or pressure. As an alternative this valve will be partial-stroke exercised quarterly and full-stroke exercised open during cold shutdown. Reverse flow closure will be demonstrated by disassembly/inspection and manually full-stroke exercising during refueling outages.

3.7.1.1.2 Evaluation--The reviewer agrees with the applicant that valve U4 008 cannot be full-stroke exercised during power operation due to the fact that the required steam flow would cause the steam driven auxiliary feedwater pump to inject cold auxiliary feedwater into the steam generator(s) which would result in thermal shock to steam generator internals. The reviewer agrees with the applicant that reverse flow closure can only be verified by disassembly/inspection.

3.7.1.1.3 Conclusion--The reviewer concludes that the applicant's proposal to partial-stroke exercise valve U4 008 quarterly, to full-stroke exercise this valve open during cold shutdown, and to disassemble this valve during refueling outages for reverse flow closure verification should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.8 Auxiliary Feedwater System

3.8.1 Category C Valves

3.8.1.1 Relief Request. The applicant has requested relief from exercising valves U4 117, 118, 119, and 120, feedwater bypass to steam generator checks, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by sample disassembly/inspection on a refueling outage frequency.

3.8.1.1.1 Applicant's Basis for Requesting Relief--These check valves cannot be exercised during power operation or cold shutdown as the only available method to verify reverse flow closure is by disassembly and observation of the disk position. As an alternative one of these valves will be disassembled and manually stroked during refueling outages on a staggered test basis. If disassembly reveals that the valve is inoperable, the remaining valves will be disassembled.

3.8.1.1.2 Evaluation--The reviewer agrees with the applicant that valves U4 117, 118, 119, and 120 can only be reverse flow closure verified by valve disassembly.

The NRC staff has concluded that valve disassembly/inspection using a manual full-stroke of the disk is an acceptable method to verify the full-stroke capability of check valves. At each disassembly the applicant must verify that the disassembled valve is capable of full-stroking and that its internals are structurally sound (no loose or corroded parts).

3.8.1.1.3 Conclusion--The reviewer concludes that the applicant's proposal to perform sample disassembly/inspection of valves U4 117, 118, 119, and 120 on a refueling outage frequency, when performed in accordance with the previous discussion (Section 3.3.1.4.2) should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.9 Condensate and Feedwater System

3.9.1 Category C Valves

3.9.1.1 Relief Request. The applicant has requested relief from exercising valves U4 071, 073, 075, and 077, feedwater checks, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by sample disassembly/inspection on a refueling outage frequency.

3.9.1.1.1 Applicant's Basis for Requesting Relief--These check valves cannot be exercised during power operation or cold shutdown as the only available method to verify reverse flow closure is by disassembly and observation of the disk position. As an alternative one of these valves will be disassembled and manually stroked during refueling outages on a staggered test basis. If disassembly reveals that the valve is inoperable, the remaining valves will be disassembled.

3.9.1.1.2 Evaluation--The reviewer agrees with the applicant that valves U4 071, C73, 075, and 077 can only be reverse flow closure verified by valve disassembly.

The NRC staff has concluded that valve disassembly/inspection using a manual full-stroke of the disk is an acceptable method to verify the full-stroke capability of check valves. At each disassembly the applicant must verify that the disassembled valve is capable of full-stroking and that its internals are structurally sound (no loose or corroded parts).

3.9.1.1.3 Conclusion--The reviewer concludes that the applicant's proposal to perform sample disassembly/inspection of valves U4 071, 073, 075, and 077 on a refueling outage frequency, when performed in accordance with the previous discussion (Section 3.3.1.4.2) should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.10 Containment Air Purification and Cleanup System

3.10.1 Category A/C Valves

3.10.1.1 Relief Request. The applicant has requested relief from exercising valves U4 001 and 002, hydrogen monitor checks, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by exercising these valves closed on a refueling outage frequency.

3.10.1.1.1 Applicant's Basis for Requesting Relief--These check valves cannot be exercised during power operation as the only method available to verify reverse flow closure is valve leak testing during Appendix J, Type C, leak testing during refue'ing outages. As an alternative reverse flow closure will be verified during Appendix J, Type C, leak testing during refueling outages.

3.10.1.1.2 Evaluation--The reviewer agrees with the applicant that valves U4 001 and 002 cannot be exercised closed during power operation or cold shutdown due to the fact that the only method available to verify reverse flow closure is valve leak testing during Appendix J Type C leak testing during refueling outages.

3.10.1.1.3 Conclusion--The reviewer concludes that the applicant's proposal to exercise valves U4 001 and 002 closed on a refueling outage frequency should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.11 Nitrogen to Accumulator System

3.11.1 Category A/C Valves

3.11.1.1 Relief Request. The applicant has requested relief from exercising valve U4 017, nitrogen supply check, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify valve operability by exercising this valve closed on a refueling outage frequency.

3.11.1.1.1 Applicant's Basis for Requesting Relief--This check valve cannot be exercised during power operation as the only method available to verify reverse flow closure is valve leak testing during Appendix J, Type C, leak testing during refueling outages. As an alternative reverse flow closure will be verified during Appendix J, Type C, leak testing during refueling outages.

3.11.1.1.2 Evaluation--The reviewer agrees with the applicant that valve U4 017 cannot be exercised closed during power operation or cold shutdown due to the fact that the only method available to verify reverse flow closure is valve leak testing during Appendix J Type C leak testing during refueling outages.

3.11.1.1.3 Conclusion--The reviewer concludes that the applicant's proposal to exercise valve U4 017 closed on a refueling outage frequency should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

3.12 Instrument Air System

3.12.1 Category A/C Valves

3.12.1.1 Relief Request. The applicant has requested relief from exercising valve U4 049, containment check, in accordance with the requirements of Section XI, Paragraph IWV-3522 and proposed to verify operability by exercising this valve closed on a refueling outage frequency.

3.12.1.1.1 Applicant's Basis for Requesting Relief--This check valve cannot be exercised during power operation as the only method available to verify reverse flow closure is valve leak testing during Appendix J, Type C, leak testing during refueling outages. As an alternative reverse flow closure will be verified during Appendix J, Type C, leak testing during refueling outages.

3.12.1.1.2 Evaluation--The reviewer agrees with the applicant that valve U4 049 cannot be exercised closed during power operation or cold shutdown due to the fact that the only method available to verify reverse flow closure is valve leak testing during Appendix J, Type C, leak testing during refueling outages.

3.12.1.1.3 Conclusion--The reviewer agrees that the applicant's proposal to exercise valve U4 049 closed on a refueling outage frequency should be sufficient to demonstrate proper valve operability. The reviewer concludes that the alternate testing proposed will give reasonable assurance of valve operability as required by the Code and, therefore, relief should be granted.

APPENDIX A

NRC STAFF POSITIONS AND GUIDELINES

APPENDIX A

NRC STAFF POSITIONS AND GUIDELINES

1. Full-Stroke Exercising of Check Valves

The NRC staff position is that check valves whose safety function is to open are expected to be full-stroke exercised. Since the disk position is not always observable, the NRC staff position is that verification of the maximum flow rate through the check valve identified in any of the plant's safety analyses would be an adequate demonstration of the full-stroke requirement. Any flow rate less than this will be considered partial-stroke exercising unless it can be shown that the check valve's disk position at the lower flow rate would permit maximum flow required through the valve. It is the NRC staff's position that this reduced flow rate method of demonstrating full-stroke capability is the only test that requires measurement of the differential pressure across the valve.

2. Valves Identified for Cold Shutdown Exercising

The Code permits valves to be exercised during cold shutdowns where it is not practical to exercise them during plant operation, and these valves are specifically identified by the applicant and are full-stroke exercised during cold shutdowns; therefore, the applicant is meeting the requirements of the ASME Code, Paragraphs IWV-3412 and -3522. Since the applicant is meeting the requirements of the ASME Code, it is not necessary to grant relief; however, during the review of the applicant's IST program, the reviewer verifies that it is not practical to exercise these valves during power operation and that the applicant's basis is valid.

It should be noted that the NRC differentiates, for valve testing purposes, between the cold shutdown mode and the refueling mode. That is, for valves identified for testing during cold shutdowns, it is expected that the tests will be performed both during cold shutdowns and each refueling outage. However, when relief is granted to perform tests on a refueling outage frequency, testing is expected only during each refueling outage. In addition, for extended outages, tests being performed are expected to be maintained as closely as practical to the Code-specified frequencies.

3. Conditions for Valve Testing During Cold Shutdown

Cold shutdown testing of valves identified by the applicant is acceptable when the following conditions are met:

a. The applicant is to commence testing as soon as the cold shutdown condition is achieved, but not later than 48 hours after shutdown, and continue until complete or the plant is ready to return to power.

b. Completion of all valve testing is not a prerequisite to return to power.

c. Any testing not completed during one cold shutdown should be performed during any subsequent cold shutdowns starting from the last test performed at the previous cold shutdown.

d. For planned cold shutdowns, where ample time is available and testing all the valves identified for the cold shutdown test frequency in the IST program will be accomplished, exceptions to the 48 hours may be taken.

4. Category A Valve Leak Test Requirements for Containment Isolation Valves (CIVs)

All containment isolation valves that are Appendix J, Type C, leak tested should be included in the IST program as Category A or A/C valves. The NRC has concluded that the applicable leak test procedures and requirements for containment isolation valves are determined by 10 CFR 50,

Appendix J. Relief from Paragraphs IWV-3412 through -3425 (1983 Edition through Summer 1983 Addenda) for containment isolation valves presents no safety problem since the intent of these paragraphs is met by Appendix J requirements, however, the applicant must comply with the Analysis of Leakage Rates and Corrective Action Requirements Paragraphs IWV-3426 and -3427 (1983 Edition through Summer 1983 Addenda). Based on the considerations discussed above, the NRC staff has concluded that the alternate testing proposed will give reasonable assurance of valve leak-tight integrity as required by the Code and that the relief thus granted will not endanger life or property or the common defense and security of the public.

5. <u>Application of Appendix J Testing to the IST Program</u>

The Appendix J review for this plant is completely separate from the IST program review. However, the determinations made by that review are directly applicable to the IST program. The applicant has agreed that, should the Appendix J program be amended, they will amend their IST program accordingly.

6. <u>Safety-Related Valves</u>

This review was limited to valves whose function is safety-related. Valves whose function is safety-related are defined as those valves that are needed to mitigate the consequences of an accident and/or to shut down the reactor to the cold shutdown conditions and to maintain the reactor in a cold shutdown condition. Valves in this category would typically include certain ASME Code Class 1, 2, and 3 valves and could include some non-Code class valves. It should be noted that the applicant may have included valves whose function is not safety-related in their IST program as a decision on their part to expand the scope of their program.

7. Active Valves

The NRC staff position is that active valves are those for which changing position may be required to shut down a reactor to the cold shutdown condition or in mitigating the consequences of an accident. Included are valves which respond automatically to an accident signal and valves which may be optionally utilized but are subject to plant operator actions, such as valves utilized to establish long term recirculation following a LOCA.

8. Rapid-Acting Power Operated Valves

The NRC staff has identified rapid-acting power operated valves as those which stroke in 2 seconds or less. Relief from the trending requirements of Section XI (Paragraph IWV-3417(a), 1983 Edition through Summer 1983 Addenda) presents no safety concerns for these valves since variations in stroke times will be affected by slight variations in the response times of the personnel performing the tests. However, the staff does require that the applicant assign a maximum limiting stroke time of 2 seconds to these valves in order to obtain this Code relief.

9. Pressurizer Power Operated Relief Valves

The NRC has adopted the position that the pressurizer power operated relief valves (PORVs) should be included in the IST program as Category B valves and tested to the requirements of Section XI. However, since the PORVs have shown a high probability of sticking open and are not needed for overpressure protection during power operation, the NRC has concluded that routine exercising during power operation is "not practical" and, therefore, not required by IWV-3410.

The PORV's function during reactor startup and shutdown is to protect the reactor vessel and coolant system from low-temperature overpressurization conditions and should be exercised prior to initiation of system conditions for which vessel protection is needed.

The following test schedule is required:

a. Full-stroke exercising should be performed at each[a] cold shutdown or, as a minimum, once each refueling cycle.

b. Stroke timing should be performed at each cold shutdown, or as a minimum, once each refueling cycle.

c. Fail-safe actuation testing should be performed at each cold shutdown.

d. The PORV block valves should be included in the IST program and tested quarterly to provide protection against a small break LOCA should a PORV fail open.

The applicant has included the PORVs (1201-PV-0455A and 0456A) in the IST program as Category B valves and the PORV block valves (HV 8000A and B) as Category B valves and is exercising them in accordance with the above guidelines.

10. Valves Which Perform a Pressure Boundary Isolation Function

The following valves meet the criteria for pressure boundary isolation valves and have been included in the IST program as Category A or A/C and are leak tested in accordance with the requirements of Section XI.

HV-8701A	
HV-8701B	RHR Pump Suction Valves
HV-8702A	
HV-8702B	

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a. The staff position described in Item A.3 regarding cold shutdown testing is not applicable to the PORVs; however, in case of frequent cold shutdowns, testing of the PORVs is not required more often than each three months.

1204	U4 120	SIS to hot leg second isolation valves
	U4 121	
	U4 122	
	U4 123	
	U6 079	Accumulator second isolation valves
	U6 080	
	U6 081	
	U6 082	
	U6 083	Injection line first isolation valves
	U6 084	
	U6 085	
	U6 086	
	U6 124	SIS to hot leg first isolation valves
	U6 125	
	U6 126	
	U6 127	
	U6 128	RHR to hot leg second isolation valves
	U6 129	
	U4 143	SIS to cold leg second isolation valves
	U4 144	
	U4 145	
	U4 146	
	U6 147	RHR to cold leg second isolation valves
	U6 148	
	U6 149	
	U6 150	

APPENDIX B

VALVES TESTED DURING COLD SHUTDOWNS

APPENDIX B

VALVES TESTED DURING COLD SHUTDOWNS

The following are Category A, B, and C valves that meet the exercising requirements of the ASME Code, Section XI, and are not full-stroke exercised every three months during plant operation. These valves are specifically identified by the owner in accordance with Paragraph IWV-3412 and 3522 and are full-stroke exercised during cold shutdowns and refueling outages. All valves in this Appendix have been evaluated and the reviewer agrees with the applicant that testing these valves during power operation is not possible due to the valve type and location or system design. These valves should not be full-stroke exercised during power operation. These valves are listed below and grouped according to the system in which they are located.

1. REACTOR COOLANT SYSTEM

1.1 Category A Valves

Residual heat removal pump suction isolation valves HV 8701A, B, 8702A, and B cannot be exercised during power operation due to a reactor coolant system pressure interlock (<750 psig) which prevents residual heat removal system overpressurization. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

1.2 Category B Valves

Reactor head vent valves HV 0442A, B, 8095A, B, 8096A, and B cannot be exercised during power operation as the downstream vent valve will open due to the reactor coolant pressure surge when exercising the upstream vent valve. This uncontrolled flow path could cause a loss of reactor coolant and system pressure. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

Power operated relief valves PV 0455A and 0456A cannot be exercised during power operation due to the resultant undesirable reactor coolant system pressure and pressurizer level transients and possible subsequent reactor trip. These valves will be full-stroke exercised during cold shutdowns and refueling outages, and as described in Section 9 of Appendix A.

2. SAFETY INJECTION SYSTEM

2.1 Category B Valves

Hot leg loop isolation valves HV 8802A and B, RWST isolation valve HV 8806, RHR to cold leg isolation valves HV 8809A and B, safety injection pump miniflow valve HV 8813, SIS cold leg injection valve HV 8835, and crossover isolation valve HV 8840 cannot be exercised due to the technical specification requirement that power is removed from the valves' operators during plant power operation. In addition, failure of these valves during testing would divert or render unavailable analyzed safety injection. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

2.2 Category A/C Valves

SIS to hot leg check valves U6 125 and 126, RHR to hot leg check valves U6 128 and 129, and RHR to cold leg check valves U6 147, 148, 149, and 150 cannot be exercised during power operation as the RHR or SI pumps cannot overcome reactor coolant operating pressure. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

3. CONTAINMENT SPRAY SYSTEM

3.1 Category B Valves

Spray additive tank outlet isolation valves HV 8994A and B cannot be exercised during power operation as the unavailability of the spray additive tank would render the containment spray system unable to perform its safety function. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

4. CHEMICAL AND VOLUME CONTROL SYSTEM

4.1 Category A Valves

RCP seal water isolation valves HV 8100 and 8112 cannot be exercised during power operation due to possible RCP seal damage. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

Charging pump to reactor coolant system isolation valve HV 8105 and letdown isolation valves HV 8152 and 8160 cannot be exercised during power operation due to interruption of pressurizer level control and possible subsequent plant shutdown. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

4.2 Category B Valves

Charging pumps to RCS isolation valve HV 8106 and letdown isolation valve HV 15214 cannot be exercised during power operation due to interruption of pressurizer level control and possible subsequent plant shutdown. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

VCT isolation valves LV 0112B and C, and RWST valves LV 0112D and E cannot be exercised during power operation because any alternate charging pump suction source would adversely affect RCS boron concentration which could result in plant shutdown. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

4.3 Category C Valves

Boric acid to charging pump check valves U4 185 and 499 cannot be exercised during power operation as this would adversely affect RCS boron concentration which could result in plant shutdown. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

5. AUXILIARY COMPONENT COOLING WATER SYSTEM

5.1 Category A Valves

Auxiliary component cooling water supply isolation valves HV 1978 and 1979 and auxiliary component cooling water return isolation valves HV 1974 and 1975 cannot be exercised during power operation because interruption of RCP thermal barrier cooling water could result in pump damage. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

5.2 Category B Valves

Thermal barrier isolation valves HV 2041, 19051, 19053, 19055, and 19057 cannot be exercised during power operation because interruption of RCP thermal barrier cooling water could result in pump damage. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

6. MAIN STEAM SYSTEM

6.1 Category B Valves

Main steam isolation valves HV 3006A, B, 3016A, B, 3026A, B, 3036A, and B cannot be exercised during power operation as the resultant severe main steam pressure transient would cause a plant shutdown. These valves will be partial-stroke exercised quarterly and full-stroke exercised during cold shutdowns and refueling outages.

Main steam power operated relief valves PV 3000, 3101, 3020, and 3030 cannot be exercised during power operation as an open failure would result in plant shutdown. These valves will be partial-stroke exercised quarterly and full-stroke exercised during cold shutdowns and refueling outages.

6.2 Category C Valves

Auxiliary feedwater pump check valves U4 006 and 404 cannot be exercised during power operation due to the resultant feedwater nozzle thermal shock when operating the steam driven auxiliary feedwater pump. This could cause steam generator feedwater nozzle cracking. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

7. AUXILIARY FEEDWATER SYSTEM

7.1 Category B Valves

Feedwater bypass isolation valves HV 15196, 15197, 15198, and 15199 cannot be exercised during power operation as the resultant interruption of feedwater flow could cause steam generator water level oscillation and subsequent reactor trip. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

7.2 Category C Valves

Auxiliary feedwater pump outlet check valves U4 001, 002, and 014, auxiliary feedwater pump suction check valves U4 013, 033, 051, 052, 058, and 061, auxiliary feedwater pump isolation check valves U4 017, 020, 023, 026, 037, 040, 043, 046, and steam generator inlet check valves U4 113, 114, 115, and 116 cannot be exercised during power operation due to the resultant feedwater nozzle thermal shock when operating the auxiliary feedwater pumps. This could cause steam generator feedwater nozzle cracking. These valves will be full-stroke exercised during cold shutdowns and refueling outages. Valves U4 051, 052, 058, and 061 will be partial-stroke exercised quarterly.

8. CONDENSATE AND FEEDWATER SYSTEM

8.1 Category B Valves

Steam generator feedwater isolation valves HV 5227, 5228, 5229, and 5230 cannot exercised during power operation as the resultant stoppage of feedwater flow would cause an undesirable steam generator water level transient and possible plant shutdown. These valves will be partial-stroke exercised quarterly and full-stroke exercised during cold shutdowns and refueling outages.

9. CONTAINMENT AIR PURIFICATION AND CLEANUP SYSTEM

9.1 Category A Valves

Purge supply isolation valves HV 2626A and 2627A and purge exhaust isolation valves HV 2628A and 2629A cannot be exercised as they have not been demonstrated capable of closing during a LOCA or steam line break accident thus technical specifications preclude opening of these valves during power operation. These valves will be full-stroke exercised during cold shutdowns and refueling outages.

10. FIRE PROTECTION WATER SYSTEM

10.1 Category A Valves

Header isolation valve HV 27901 cannot be exercised during power operation as failure in the open position would unnecessarily compromise containment integrity. This valve will be full-stroke exercised during cold shutdowns and refueling outages.

11. INSTRUMENT AIR SYSTEM

11.1 Category A Valves

Isolation valve HV 9378 cannot be exercised during power operation as failure in the closed position would case a loss of containment instrument air. The resultant loss of plant letdown capability could cause loss of pressurizer level control and subsequent plant shutdown. This valve will be full-stroke exercised during cold shutdowns and refueling outages.

APPENDIX C

P&ID LIST

APPENDIX C

The P&IDs and drawings listed below were used using the course of this review.

System	Drawing No.	Revision
Post Accident Sampling	1X4DB110	4
Reactor Coolant	1X4DB111	10
Reactor Coolant	1X4DB112	13
Chemical and Volume Control	1X4DB114	12
Chemical and Volume Control	1X4DB116-1	6
Chemical and Volume Control	1X4DB116-2	6
Chemical and Volume Control	1X4DB118	9
Safety Injection	1X4DB119	10
Safety Injection	1X4DB120	8
Safety Injection	1X4DB121	11
Residual Heat Removal	1X4DB122	11
Waste Processing-Liquid	1X4DB127	9
Spent Fuel Cooling and Purification	1X4DB130	12
Containment Spray	1X4DB131	11
Nuclear Service Cooling Water	1X4DB133-1	11
Nuclear Service Cooling Water	1X4DB133-2	12
Nuclear Service Cooling Water	1X4DB134	10
Nuclear Service Cooling Water	1X4DB135-1	12
Nuclear Service Cooling Water	1X4DB135-2	11
Component Cooling Water	1X4DB136	10
Auxiliary Component Cooling Water	1X4DB138-1	9
Auxiliary Component Cooling Water	1X4DB138-2	10

System	Drawing No.	Revision
Nuclear Sampling - Liquid	1X4DB140	9
Containment and Auxiliary Building Drains - Radioactive	1X4DB143	13
Main Steam	1X4DB159-1	13
Main Steam	1X4DB159-2	11
Main Steam	1X4DB159-3	7
Auxiliary Feedwater	1X4DB161-2	10
Auxiliary Feedwater Pump	1X4DB161-3	9
Condensate and Feedwater	1X4DB168-3	12
Diesel Generator	1X4DB170-1	8
Diesel Generator	1X4DB170-2	7
Fire Protection Water	1X4DB174-4	9
Service Air	1X4DB186-1	10
Instrument Air	1X4DB186-2	10
Plant Demineralized Water	AX4DB190-2	10
Purification and Clean-Up	1X4DB213-1	4
Purification and Clean-Up	1X4DB213-2	3
Safety Related Chillers	1X4DB221	9

APPENDIX D

IST PROGRAM ANOMALIES IDENTIFIED DURING THE REVIEW

APPENDIX D

IST PROGRAM ANOMALIES IDENTIFIED DURING THE REVIEW

Inconsistencies and omissions in the applicant's program noted during the course of this review are summarized below. The applicant should resolve these items in accordance with the evaluations, conclusions, and guidelines presented in this report.

1. Pump relief request No. 2 requests relief from measuring boric acid transfer pumps flow due to flow instruments not being installed. Not having instrumentation installed does not negate the requirement to measure flow (section 2.2.1 of this report).

2. Valve relief request No. 1 requests relief from the corrective action requirement of measuring degraded and increasing valve stroke times monthly for valves identified as being tested on a cold shutdown frequency (Appendix B of this report). The applicant wishes to measure cold shutdown frequency tested degraded valves not being repaired as the monthly testing would not be applicable and the plant could go to power operation with the degraded valves. Relief should not be granted as corrective action requires the degraded valves to be repaired (Section 3.1.3 of this report).

APPENDIX V

TECHNICAL EVALUATION REPORT OF THE
SAFETY PARAMETER DISPLAY SYSTEM
VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

TECHNICAL EVALUATION REPORT

OF THE

GEORGIA POWER COMPANY

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

SAFETY PARAMETER DISPLAY SYSTEM

February 17, 1987

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for the

United States

Nuclear Regulatory Commission

TECHNICAL EVALUATION REPORT
FOR THE
GEORGIA POWER COMPANY
VOGTLE ELECTRIC GENERATING PLANT, UNIT 1
SAFETY PARAMETER DISPLAY SYSTEM

1. INTRODUCTION

NUREG-0660 (1) identified the need for power reactor licensees and applicants for operating licenses to provide a Safety Parameter Display System (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This need was confirmed by NRC in NUREG-0737 (2) and Supplement 1 to NUREG-0737 (3). SPDS requirements in Supplement 1 to NUREG-0737 replaced those in earlier documents.

Included in Supplement 1 to NUREG-0737 is the requirement that the licensee or applicant prepare a written safety analysis for the SPDS and provide this analysis along with the plant-specific SPDS implementation plan for NRC review. Criteria for evaluating Safety Parameter Display Systems are contained in Section 18.2 of NUREG-0800 (4), the Standard Review Plan. These criteria address both the review of a specific SPDS design and review of the applicant's or licensee's verification and validation (V&V) program, including the program for SPDS design, development, and testing. Results of the NRC evaluation of a SPDS will be documented in a Safety Evaluation Report (SER) or SER Supplement.

This Technical Evaluation Report (TER) provides Lawrence Livermore National Laboratory's (LLNLs) evaluation of the Vogtle Electric Generating Plant's SPDS with respect to the requirements of Supplement 1 to NUREG-0737, for NRC's use in preparing a Safety Evaluation Report. This TER is based upon examination of the V&V program, review of the operation of the SPDS, review of Georgia Power Company's Safety Analysis Report for the SPDS, additional material submitted by Georgia Power Company (GPC), documentation of meetings between GPC and the NRC, and an on-site audit conducted on December 3 and 4, 1986. The audit specifically addressed the points of both a Design Verification Audit and a Design Validation Audit as described by Sec. 18.2 of NUREG-0800 (4). The Audit Team was composed of one individual from the NRC and two individuals from LLNL, acting as consultants to the NRC.

2. SAFETY PARAMETER DISPLAY SYSTEM DESIGN OVERVIEW

The SPDS at Vogtle is a function of the Emergency Response Facility Computer System (ERFCS). The SPDS function is provided by a set of displays depicting seven critical safety functions (CSFs). These CSFs are Reactivity, Core Cooling, Heat Sink, RCS Integrity, Containment Conditions, RCS Inventory, and Radiation Monitoring. There are three levels of displays for CSFs.

> Top-level displays showing the value of specific top-level parameters, in the form of color coded deviation bar charts or tabular displays and status of engineered safeguards and CSF status.

Second-level displays depicting a logic path for each Critical Safety Function Status Tree (CSFST) (a CSFST corresponds to each CSF except Radiation Monitoring). A logic path indicates whether a CSF is satisfactory or challenged, and the basis for determining the degree of challenge. For Radiation Monitoring the second-level displays consist of tabular presentation of radiation monitoring inputs.

Third-level displays include time-history and parameter vs. parameter plots, numeric indication of CSF parameter values, and status indication for SPDS discrete inputs.

Color coding of continuously-displayed CSF status boxes and logic trees is used to indicate the status of each CSF. Visual alarms (status box flashing and appropriate color coding of status boxes, logic trees, etc.) are presented to alert operators that alarm threshold values have been exceeded. Auditory alarms are presented to indicate that CSF status changes have occurred. CSFSTs are based upon plant-specific Emergency Operating Procedures (EOPs) that were derived from Westinghouse Owners' Group (WOG) upgraded Emergency Response Guidelines (ERGs). Data updates occur every 2.5 seconds, except for radiation monitoring data. Radiation monitoring data is updated every 5 seconds when the primary communications link between the Plant Effluent Radiation Monitoring System (PERMS) and the SPDS is functional. When the backup PERMS/SPDS communications link is functional this update rate can extend to once every 30 seconds. The time scale on trend displays is user selectable; the default trend display time scale is 2.5 seconds.

User access to CSF-related displays is via a keypad configured to correspond top-to-bottom to the hierarchy of CSF importance and, except for radiation monitoring, in a left-to-right fashion corresponding to the display hierarchy. Below the keys which control access to CSF-related keys is a grouping of keys which provide access to Radiation Monitoring second- and third-level displays. A single key press will access any SPDS display. System response time for user interactions is less than 3 seconds.

SPDS displays can be called up on any of three SPDS terminals in the Control Room, five SPDS terminals in the Technical Support Center, and four terminals in the Emergency Operations Facility (EOF).

Analog inputs from several plant systems, including the Plant Safety Monitoring System (PSMS) and the Plant Effluent Radiation Monitoring System (PERMS), are received by the ERFCS. Data validity checks, based both upon the range of each instrument and interchannel comparisons of redundant inputs, are performed. Data which are determined to be "bad" (invalid) are not displayed, and data which are determined to be questionable are distinguished from good data through use of color coding. Neither "bad" nor questionable data are used in the determination of CSFSTs.

3.0 ASSESSMENT OF THE VERIFICATION AND VALIDATION PROGRAM

A V&V program is concerned with the process of specification, design, fabrication, testing, and installation associated with an overall system's software, hardware, and operation. For the SPDS, verification is the review of the requirements to see that the right problem is being solved, review of the design to see that it meets the requirements, and testing of system modules to verify that they function properly. Validation includes performance testing of the integrated system to see that it meets all requirements. Validation testing should not only include integrated testing of the hardware and software, but testing of the SPDS as part of the larger system for plant operations which includes the control room, plant procedures, plant operators, and operator training.

Supplement 1 to NUREG-0737 does not require that Verification and Validation of the SPDS be conducted. However, a V&V program performed by the applicant/licensee during design, installation and implementation of an SPDS will facilitate the NRC review of the system. On the basis of an effective V&V program, the NRC staff will reduce the scope and detail of the technical audit of the design.

This section presents LLNL's assessment of the V&V program. The criteria for an effective V&V program recommended by Section 18.2 of NUREG-0800 and by NSAC/39 were used as the basis of this assessment.

3.1 SYSTEM REQUIREMENTS REVIEW

The system requirements are the foundation on which the completed system must be designed, built, and accepted. Section 18.2 of NUREG-0800 recommends that a review of system requirements be conducted to determine that the SPDS functional needs will be satisfied. NSAC/39 states that a system requirements review should independently determine if the requirements will result in a possible and usable solution to the entire problem, and should verify that the requirements are correct, complete, consistent, understandable, feasible, testable, and traceable.

3.1.1 Discussion

Formal functional requirements for the ERFCS were developed by GPC. However, because the initiation of ERFCS design preceded the issuance of NRC guidance on V&V, these formal requirements were not prepared before initiation of detailed system design. Therefore, formal functional requirements were not used to guide SPDS design and coding activity.

A system requirements document was prepared by GPC for the V&V activity. This document was reviewed for GPC by Energy Incorporated (EI) to verify that the ERFCS design satisfies pertinent NRC requirements. The bases for the SPDS portion of this review were developed from NUREG-0660, NUREG-0696, NUREG-0700, NUREG-0737, Supplement 1 to NUREG-0737, and NUREG-0835. The methodology and results of the system requirements review are documented in a system requirements review report (10). GPC either corrected deficiencies noted in this report, or presented a rationale for not correcting such deficiencies. Finally, GPC committed to use the system requirements as a basis for future SPDS modifications.

The NRC Audit Team examined the functional requirements document and the system requirements review report. Sample SPDS design criteria from Supplement 1 to NUREG-0737 and NUREG-0800 were traced to the functional requirements document using the system requirements review report. The sample design criteria selected were found to be appropriately addressed in the requirements document, and adequately reviewed during the system requirements review.

3.1.2 LLNL Evaluation

GPC has adequately addressed this recommendation of Section 18.2 of NUREG-0800.

3.2 DESIGN VERIFICATION REVIEW

Section 18.2 of NUREG-0800 recommends that a design verification review be performed after the system is initially designed to verify that the design will satisfy functional needs. NSAC/39 recommends that the design review ensure that the system requirements decomposition into hardware and software is complete, and that there are no ambiguities or deficiencies.

3.2.1 Discussion

The Vogtle ERFCS hardware design was procured in accordance with a specification prepared for GPC by Bechtel Power Corporation. ERFCS software was developed via an informal process that did not entail preparation of software requirements documents.

A design verification review was conducted for GPC by EI. This review was performed after delivery of ERFCS hardware, and after most software development was complete. EI reviewed the ERFCS hardware specification and the design files against the requirements stated in the functional requirements document. Traceability matrices were developed to document that each functional requirement had been incorporated into the hardware specification or the software design, as appropriate. EI also reviewed selected portions of the ERFCS program to verify that planned software features were appropriately translated into code. Deficiencies identified by this review were either corrected by GPC, or justification for not correcting the deficiency was documented.

EI reviewed approximately 10 percent of the SPDS software design to verify:

Conformance of the design to system requirements.

Adequacy of design documentation.

Adequacy of programming.

Display design.

Requirements were reviewed by conducting a traceability analysis. The software design and implementation was deemed acceptable by EI.

GPC also performed a V&V test to ensure that the software algorithms are in conformance to functional descriptions. Modes and inputs were randomly selected, and every combination of the redundant input matrix was tested. Each verification was signed off individually.

The NRC Audit Team examined the design review report (13). This effort confirmed that sample functional requirements were addressed by the design review and were traceable to the hardware specification and the software design files.

Verification of the human factors design of the ERFCS user interface was performed by GPC as part of the Detailed Control Room Design Review (DCRDR). This review evaluated the SPDS consoles and displays with respect to the criteria of NUREG-0700, Section 6. This review also confirmed that color coding conventions, nomenclature, and abbreviations used in ERFCS displays are consistent with those used in the rest of the control room. Human engineering discrepancies (HEDs) identified by this review were assessed for correction as part of the DCRDR process.

The PSMS is the subject of a generic V&V Program, which is under separate review by the NRC. Any differences between the generic PSMS and the PSMS at Vogtle will be investigated as part of a supplemental V&V effort. The supplier of PERMS conducted V&V testing on PERMS. Signal reference inputs were tested for both parameter value and status word output. The test report is not part of GPC's documentation, but is available for review at the vendor. GPC has not reviewed the PSMS or PERMS V&V activities to confirm that all system characteristics that are important to the SPDS functions have been appropriately verified and validated.

The NRC Audit Team reviewed the Control Room Survey Checklists applicable to the SPDS. This review confirmed that the checklists were appropriately completed, and that HEDs were appropriately recorded. GPC's process for assessing and correcting HEDs was previously reviewed and found acceptable by NRC.

Future modifications to ERFCS software will be controlled by a plant procedure (20). The NRC Audit Team examined this procedure, and confirmed that it includes provisions requiring design review and verification testing of changes. Furthermore, this procedure requires a periodic verification of the existing software to ensure computer program integrity. Although a satisfactory design verification review was completed, the SPDS development process at Vogtle would have benefited significantly from the development of software requirements to guide the software development process. The development of these requirements in a largely post hoc fashion resulted in minimal impact of these requirements upon system development.

3.2.2 LLNL Evaluation

With respect to the SPDS functions performed by the ERFCS, GPC has adequately addressed the recommendations of Section 18.2 of NUREG-0800 concerning design verification review. In order to completely address the need for SPDS design verification, GPC must complete design verification activity with respect to the entire SPDS, including completion of PSMS V&V, and confirmation that PSMS and PERMS V&V adequately address the SPDS support functions of these systems. Prior to the first refueling, GPC should report to NRC on the completion of this confirmation.

3.3 VALIDATION TESTING

NUREG-0800, Section 18.2 recommends that validation testing be performed after the system is assembled to confirm that the operating system satisfies functional needs.

3.3.1 Discussion

GPC's validation process included man-in-the-loop testing conducted as part of the EOP validation exercises. These exercises tested each operator task described by the EOPs, including tasks included in the functional response guidelines. The validation exercises were conducted in the Vogtle control room simulator, which included a functioning SPDS. During the course of each exercise, the SPDS was used by a Shift Technical Advisor (STA) to monitor the status of CSFs. Following each exercise, a debriefing was held to discuss problems encountered during the drill. The results of the post-exercise debriefing were documented on validation comment sheets and debriefing questionnaires.

The NRC Audit Team examined the Debriefing Questionnaires and validation comment sheets for two EOP validation scenarios (11,12). This examination showed that neither the comment sheets nor the questionnaires specifically prompted participants for comments on the SPDS. Nevertheless, comments on the SPDS were provided by the primary user. The Audit Team was unable to verify that the comments provided were adequately addressed.

Validation testing included system response time tests. These tests confirmed that under maximum system loading, display call-up time and screen update rates are under three seconds for all parameters, except those provided to the ERFCS by the Radiation Monitoring System. The slow response for RMS parameters is identified by GPC as a deficiency to be corrected.

Finally, GPC conducted integrated systems validation testing. Most of this testing occurred in GPC's Atlanta offices, using Vogtle Unit 2 hardware (which is identical to Unit 1 hardware). Some testing was repeated on site using Unit 1 hardware. The integrated systems validation testing confirmed that system performance meets functional system specifications. Difficulties were noted, then were promptly resolved. Appropriate test documentation was maintained throughout this entire activity, which was reviewed by EI.

3.3.2 LLNL Evaluation

GPC has satisfied the intent of this recommendation of NUREG 0800, Section 18.2 with respect to integrated hardware/software system testing. With respect to the broader concept of validating the SPDS design in the context of the Vogtle control room, procedures, and operator training, GPC has only partially addressed this recommendation. The man-in-the-loop testing conducted by GPC was adequate to demonstrate that the SPDS displays and CSF evaluation logic can be used to evaluate the status of plant safety functions. This testing did not, however, validate the CSFST logic used during normal operation, nor did it prompt validation test participants to identify improvements that should be made to the SPDS user interface. Therefore, the EOP validation testing does not constitute a complete and rigorous man-in-the-loop test of the SPDS. To compensate for this shortcoming, GPC should actively solicit operator feedback on the usability of the SPDS after sufficient experience has been gained to make operators' comments meaningful. The process used in acquiring this feedback should prompt the users for comments on the normal operation status trees and specific SPDS design features. Furthermore, the feedback process should include solicitation of operators' opinions regarding the potential enhancements noted by the NRC Audit Team and described in Section 4.9.1 of this TER.

GPC should provide for NRC review a description of the process and results of operator feedback resulting from experience in using SPDS. This description should be submitted by first refueling.

3.4 FIELD VERIFICATION TESTS

NUREG-0800, Section 18.2 recommends performance of field verification tests, once the system is installed, to verify that the validated system was installed properly. NSAC/39 recommends that, as a minimum, field verification testing should confirm that the information displayed is directly correlated with the sensor data being input.

3.4.1 Discussion

GPC reported that end-to-end channel tests were conducted with the installed SPDS system. These tests verified that the current value of each instrument input was accurately stored by the ERFCS, and that displayed data correspond to sensor data. In addition, normal periodic instrument loop calibration includes verification that sensor data are properly displayed on the SPDS.

3.4.2 LLNL Evaluation

GPC has adequately implemented the recommendations of Section 18.2 of NUREG-0800 regarding field verification testing.

4. ASSESSMENT OF SPDS DESIGN

4.1 "THE SPDS SHOULD PROVIDE A CONCISE DISPLAY"

4.1.1 Discussion

The Vogtle SPDS provides an overview of the status of the seven CSFs. This overview is in the form of seven appropriately labeled status boxes. Color coding for the status boxes as well as tabular/graphic data and logic path diagrams is according to the following conventions:

- Red - extreme challenge to CSF
- Orange - severe challenge to CSF
- Yellow - alert condition
- Green - satisfactory condition

The status boxes are part of all first and second level displays. The first level displays provide an overview of SPDS parameters. Users may select either a numeric-tabular or deviation bar graph format for the presentation of first and second level displays. Questionable data are displayed in magenta. Bad data are not displayed; instead, "BAD" appears in magenta where numerical values normally appear.

The second level displays indicate individual CSFSTs for all but the Radiation Monitoring CSF. These CSFSTs depict whether a CSF is satisfactory or is challenged, and, if a CSF is challenged, the specific data used in determining the degree of challenge. The second level Radiation Monitoring CSF display depicts a map and status of radiation levels in every area of the plant. Bad or questionable data are displayed as previously described, and status information data for CSFs that cannot be evaluated are displayed in magenta.

The Control Room SPDS workstation contains two displays. The left visual display terminal (VDT) is used only for the presentation of first and second-level displays. The right VDT allows users to access first, second, and third-level displays, as well as other ERFCS displays. Third-level displays consist of time history plots of SPDS parameters, parameter vs. parameter plots, and numerous tabular displays yielding detailed information about plant status. CSF status boxes are not included on third-level or other ERFCS displays.

Status information for SPDS parameters is available at a single workstation. Displays are appropriately organized and formatted to facilitate comparison of data from related plant functions and assessment of plant safety status. Appropriate use of color and configural coding enhances perception of critical plant parameters.

4.1.2 Assessment

The Vogtle SPDS meets the requirements of Supplement 1 to NUREG-0737 regarding concise display of CSFs.

4.2 "THE SPDS SHOULD ... DISPLAY ... CRITICAL PLANT VARIABLES"

4.2.1 Discussion

The SPDS parameters selected for display and the groupings of parameters into CSFs are based upon the CSFs monitored by the Westinghouse upgraded EOPs. The parameter groupings are:

Reactivity.

Core cooling.

Heat sink.

Reactor cooling system integrity.

Reactor cooling system inventory.

Containment.

Parameters used to assess the CSF Status Trees are as follows. Some parameters are not used during all plant operating modes.

Reactivity

- o Power range neutron flux
- o Intermediate range neutron flux
- o Intermediate range startup rate
- o Source range startup rate
- o Source range detector voltage
- o Reactor cooling system average temperature

Core Cooling

- o Core exit temperature
- o Reactor cooling system subcooling
- o Reactor coolant pump status
- o Reactor vessel level, full range
- o Reactor vessel level, dynamic head range
- o Residual heat removal pump status
- o Residual heat removal valve positions

Heat Sink

- o Steam generator wide range level
- o Steam generator narrow range level
- o Auxiliary feedwater flow
- o Steam generator pressure
- o Residual heat removal pump status
- o Residual heat removal valve positions

RCS Integrity

- o RCS cold leg temperature
- o RCS wide range pressure
- o Power operated relief valve positions

Containment

- o Containment pressure
- o Containment water level
- o Containment radiation
- o Containment temperature
- o Containment hydrogen concentration (displayed by SPDS but not included in Containment CSFST assessment)

RCS Inventory

- o Pressurizer level
- o Reactor vessel level, head upper range
- o Reactor vessel level, full range level

In addition to the safety functions defined by the Westinghouse ERGs, the Vogtle SPDS monitors the status of the Radiation Control CSF. The status determination for this CSF is based upon monitoring a large number of plant radiation monitoring channels including:

- o Main Steam Line Radiation
- o Plant Vent Radiation
- o Containment Area Radiation

The above groups encompass the five safety functions listed in Supplement 1 to NUREG-0737.

4.2.2 LLNL Evaluation

The parameters displayed by the Vogtle SPDS are sufficient to provide operators with information regarding the status of the five safety functions identified by Supplement 1 to NUREG-0737.

4.3 "THE SPDS SHOULD ... AID THEM (OPERATORS) IN RAPIDLY AND RELIABLY DETERMINING THE SAFETY STATUS OF THE PLANT"

4.3.1 Discussion

As mentioned in 2.0, parameter values displayed by SPDS and used by SPDS logic trees are updated every 2 1/2 seconds, even when the demand level on the ERFCS computer is relatively high. The exception is PERMS data updates, which can occur as slowly as every minute. However, GPC has committed to reducing the PERMS update rate to 10 seconds or less. The update rate for trend plots may be slower if the user selects a longer trending interval. System response time for user interaction is consistently less than 3 seconds.

SPDS parameters originate mainly from PERMS and PSMS. These computers receive analog sensor data, then transform these inputs to digital data in engineering units. The PERMS performs validity checks (based on calculated range for each instrument) for each radiation monitoring data channel, then sends parameter values and status flags to the ERFCS. One flag designates channel status (test, inactive, or trouble). The other flag denotes alarm status; alarms result when a signal exceeds an alarm set point. PSMS checks inputs for operability and against the possible input range, and passes parameter value and status information to the ERFCS. The ERFCS interprets any data which are not flagged "good" as "bad" data.

Once data inputs are received from PERMS and PSMS, the ERFCS also performs two types of data validity checks:

Range checks. Input is checked to ensure that it falls within a range of data values for the instrument from which the input was obtained. GPC stated that input range checking criteria generally reflect the limits of instrument capabilities rather than realistic operating ranges. However, individual exceptions have been made for some instruments.

Interchannel comparison checks. This comparison is performed for all parameters that have more than one input. If no good inputs are available for a parameter, that parameter is labeled "bad." If all inputs are good, and if they are all within a delta of the average of all inputs, then the average value is displayed and flagged as valid. Otherwise, the average value of all inputs that are within delta of the average is displayed, and the value is flagged questionable.

Delta for the interchannel comparison check is currently defined to be two percent of the input instrument range for each parameter. The NRC Audit Team pointed out that under adverse environment conditions, instrument accuracy is likely to deteriorate, such that valid readings may differ by more than 2 percent. Therefore, the choice of an arbitrary value of delta is inappropriate. Two sets of deltas, one for normal operating conditions, and one for adverse environment conditions, may be warranted. In reference 22 GPC committed to review the selection of interchannel comparison validity criteria and implement any needed revisions prior to attaining five percent thermal power. SPDS users receive feedback about whether or not SPDS is functional. A watchdog timer in the ERFCS determines whether screens are refreshed within a criterion time period. If a screen refresh does not occur within this time period, the screen blanks, and an alarm is presented. This refresh check is performed locally in the VDT.

As mentioned previously, GPC has conducted end-to-end channel tests to verify that displayed data correspond to inputs from instruments. GPC has also designed SPDS software to compensate for problems such as disc failure. As part of GPC's preventative maintenance program, periodic instrument loop calibration will verify that sensor data correspond to displayed data values. Periodic testing will also be performed to verify that data handling routines of the software function properly.

The NRC Audit Team examined sample ERFCS software. Algorithms for pressurizer pressure engineering units conversion and data validation were analyzed. In a line-by-line walkthrough, GPC showed that the software algorithm which determines pressurizer pressure values functions as described by functional descriptions. Engineering units conversion is, in the case of pressurizer pressure, accomplished through use of linear conversion of voltage inputs. The NRC Audit Team verified that this conversion is appropriate. Finally, GPC demonstrated that software algorithms to produce data validation outputs conform to functional descriptions provided by GPC.

GPC has not yet evaluated the availability of SPDS. However, GPC has committed to test system availability, using a 1000-hour availability test. This test will address the availability of SPDS as a system including the ERFCS, the power supply, and systems such as PERMS and PSMS which provide data to ERFCS. Repair times used in the availability determination will consider GPCs spare parts stocking plans and maintenance staffing levels.

SPDS displays numeric values to the nearest 1/10 or nearest 1/1000, depending upon the scale used. Trend plots are displayed with single pixel accuracy. Trend plot parameter value scales are auto-ranged to display the plot on the largest scale that will accommodate the range of the data.

According to GPC, system security is accomplished primarily through limiting access to terminal keyboard function cards that allow data and software changes. Plant personnel may have access to virtually any VDT, but cannot change system information. The cards which allow input of mode changes (which affect set points for alarms) are locked in the shift supervisor's desk. A programmer's card is necessary to modify system software. This card is stored in a secure area. Then access to the programmer's card must be installed in the control room VDT and system operations changes must be entered in a different VDT if software modifications are to be made. Finally, passwords are required to make these modifications. As a final safeguard, only 15 minutes system access is allowed by the system.

4.3.2 LLNL Evaluation

The Vogtle SPDS will satisfy the provisions of Supplement 1 to NUREG-0737 regarding rapid and reliable display of SPDS information once:

- GPC has established and implemented realistic rather than arbitrary criteria for interchannel comparison of redundant inputs. These values must be appropriate both for adverse and normal operating conditions, and must be based on anticipated instrument loop accuracies.

- Acceptable system availability has been demonstrated.

GPC should describe to the NRC the results of activities undertaken to address these issues. This information should be submitted no later than first refueling.

Additionally, LLNL suggests that range checking criteria should be modified to reflect the actual operating range rather than the upper-lower limit for each instrument.

4.4 "THE PRINCIPAL PURPOSE AND FUNCTION OF THE SPDS IS TO AID THE CONTROL ROOM PERSONNEL DURING ABNORMAL AND EMERGENCY CONDITIONS IN DETERMINING THE SAFETY STATUS OF THE PLANT AND IN ASSESSING WHETHER ABNORMAL CONDITIONS WARRANT CORRECTIVE ACTIONS BY CONTROL ROOM OPERATORS TO AVOID A DEGRADED CORE."

4.4.1 Discussion

The Vogtle SPDS displays the current value of input variables, and provides perceptual cues to abnormal values through use of status color coding, as described previously in Section 4.1.1. This color coding enables users to quickly determine the status of each CSF, the basis for the SPDS's status determination, and which plant parameters related to CSF status deviate from normal. Primary SPDS displays depict SPDS variables in either tabular or bar chart format, both of which are user selectable. Additionally, an audible alarm is presented to call users' attention to any change in CSF status other than a change to satisfactory status.

The Vogtle SPDS is capable of displaying historical trends for any variable input to the ERFCS, including all SPDS variables. A 10 minute trending interval is displayed by default, and time resolution of trend plots is 1/60 of the selected trending interval. During validation testing, GPC determined that this default time base yields the highest resolution of the variable and time scale. Additional trending intervals up to 2 hours are available. Trend displays are auto-ranged, such that the size of the display is adjusted so that the plot fills the screen.

The variables displayed, logic, logic set points, and logic path formats are based on the CSF evaluation processes contained in the Vogtle EOPs, which are based on the ERGs and Functional Response Guidelines (FRGs) developed for WOG. In fact, for post-trip conditions second-level SPDS displays are identical to the FRG status trees. Because WOG Guidelines are based on a system function and task analysis, CSF displays are therefore traceable to a system and task analysis.

GPC also developed specific CSF status trees for each plant operating mode. This feature enhances the usefulness of CSFSTs during normal operation. The normal mode trees are based upon the post-trip condition with changes of decision point values and logic as needed to reflect normal conditions. These changes were based upon review of Technical Specification Limiting Conditions for Operation and System Function Analysis.

4.4.2 LLNL Evaluation

The Vogtle SPDS adequately provides the operator aid in the determination of safety status, and therefore, fulfills this requirement of Supplement 1 to NUREG-0737.

4.5 "THE SPDS (SHALL BE) LOCATED CONVENIENT TO THE CONTROL ROOM OPERATORS"

4.5.1 Discussion

Three Vogtle SPDS VDTs are located in the control room. Two are located near instrument boards, and one is located at the shift supervisor's console. There is an ERFCS console to the rear of the chief operator's console, and one of these consoles is dedicated to SPDS. The SPDS does not interfere with operator movement; there is an aisleway between other consoles and SPDS VDTs.

The shift technical advisor has been designated as the primary SPDS user under adverse plant conditions. GPC stated that this person is an integral part of the operating staff, and that, accordingly, there is a normal duty station for this person in the control room.

4.5.2 LLNL Evaluation

GPC has met the requirement of Supplement 1 to NUREG-0737 that the SPDS be convenient to operators.

4.6 "THE SPDS SHALL CONTINUOUSLY DISPLAY INFORMATION FROM WHICH THE SAFETY STATUS OF THE PLANT...CAN BE ASSESSED..."

4.6.1 Discussion

As mentioned previously, all top- and second-level SPDS displays include color-coded boxes that indicate the current status of each CSF. Additionally, summary overviews of the status are available. User selectable, top-level deviation-bar-chart and tabular displays provide users with additional overview of important plant parameters. Also mentioned previously is that the Vogtle SPDS provides appropriate perceptual cues and configural displays to facilitate users' ability to determine overview safety status information.

One of the three VDTs in the control room is dedicated to the display of top and second-level SPDS information. If the ERFCS detects failure in one of the control room consoles, a software interlock ensures that top level displays will appear on one VDT in the control room.

4.6.2 LLNL Evaluation

The Vogtle SPDS satisfies the requirement of Supplement 1 to NUREG-0737 that the SPDS shall continuously display information from which the safety status of the plant can be determined.

4.7 "THE SPDS SHALL BE SUITABLY ISOLATED FROM ELECTRICAL OR ELECTRONIC INTERFERENCE WITH EQUIPMENT AND SENSORS THAT ARE IN USE FOR SAFETY SYSTEMS"

4.7.1 Discussion

GPC indicated that Class 1E isolation devices are used at each interface between Class 1E systems and the SPDS. Test type data for the specific isolation devices has been separately provided to the NRC.

4.7.2 LLNL Evaluation

Review of the isolation provisions is not within the scope of this Technical Evaluation Report.

4.8 "PROCEDURES WHICH DESCRIBE THE TIMELY AND CORRECT SAFETY STATUS ASSESSMENT WHEN THE SPDS IS AND IS NOT AVAILABLE WILL BE DEVELOPED BY THE LICENSEE IN PARALLEL WITH THE SPDS. FURTHERMORE, OPERATORS SHOULD BE TRAINED TO RESPOND TO ACCIDENT CONDITIONS BOTH WITH AND WITHOUT THE SPDS AVAILABLE."

4.8.1 Discussion

The Functional Response Guidelines of the Vogtle Emergency Operating Procedures are used in the determination of safety status. The Vogtle SPDS in essence provides an automated means to continuously evaluate the CSFSTs contained in the plant EOPs. If the SPDS is unavailable, hardcopies of FRGs are available for operators to use without the aid of automation.

Operator training in the use of the SPDS is incorporated into training in the use of plant EOPs. This training, required for operator licensing and requalification, includes use of EOPs with and without the SPDS.

4.8.2 LLNL Evaluation

The Vogtle SPDS meets this requirement of Supplement 1 to NUREG-0737.

4.9 "THE SPDS DISPLAY SHALL BE DESIGNED TO INCORPORATE ACCEPTED HUMAN FACTORS PRINCIPLES SO THAT THE DISPLAYED INFORMATION CAN BE READILY PERCEIVED AND COMPREHENDED BY SPDS USERS."

4.9.1 Discussion

The logic path formats of the CSFSTs were developed by Westinghouse using their human factors design criteria and input from utility representatives participating in WOG. GPC maintained that there has been substantial integration of human factors principles into SPDS, especially with respect to the design of the information hierarchy and the development of redundant formats and redundant methods to access information. The lead human factors engineer and the Control Room Design Review Team provided this input.

The NRC Audit Team observed that the SPDS is adequate from the perspective of user interface design. The keypad layout is based on the relative priority of the CSFs as defined by the Vogtle EOP FRGs. Traversals from one function to another are rapid and simple, and it is virtually impossible for users to become lost in the system while attempting such traversals. Users are generally provided with appropriate feedback after making traversal responses, e.g., "Historical Data Collection in Progress." Color coding is appropriate, and the number of colors presented is well within the limits of human coding abilities.

Although the SPDS user interface design is adequate, the NRC Audit Team noted a number of areas in which design improvements would be likely to enhance SPDS usability.

- In the Vogtle color coding scheme, perceptual cues for challenges to CSFs are lost when a CSF parameter is of questionable validity. This occurs because the color coding to indicate questionable validity takes priority over color coding of parameter alarm status. Vogtle's data validity criteria are so stringent, however, that data identified as questionable will often be valid. The operators may find the system to be more useful if some other cue is provided to indicate questionable data, and the color coding to indicate alarm status is maintained.

- The containment isolation valve status display uses the color codes red for open and green for closed. This is consistent with the convention for valve position lights, but is not consistent with the SPDS convention of green for safe, red for unsafe. Conversely, use of the green/safe, red/unsafe convention would violate the valve status color convention. Operator input would be useful in determining which convention violation causes the least confusion.

- Parameter alarm status is shown as green for normal, red for high, and flashing red for high-high or low-low. Parameter alarm color coding might be more easily understandable if the CSF color coding scheme of green-normal, yellow-alert, orange-severe challenge and red-unsafe is used.

- Few prompts are currently presented. Required user responses might be less ambiguous if prompts were used to guide parameter value selection with keyboard arrow keys, and to guide numerical inputs via keyboard.

- The color of indicated set points and data plots is sometimes the same, making discrimination difficult or impossible. A change in color utilization should be considered.

- Acceptable operating levels are often not indicated on graphic displays. Indication of these levels might be useful to SPDS users.

- Default values are generally not presented. Specification of options and input values would probably be easier if such values were indicated.

- Sometimes the underline cursor which is displayed is difficult to locate. A block cursor might alleviate this problem.

- Displays may contain numerous numerical values, some of which may be selected to bring up additional data screens, and some of which may not. Differential coding of selectable and non-selectable values would probably help users avoid erroneous selections.

- Indication of current parameter values on status tree displays might be useful information for users.

- The cursor often homes in a location from which it must be moved for data input or selection of options. Unnecessary, additional interaction steps could be eliminated if the cursor would home in an active data input or option selection area.

- Scroll keys would be easier to use if the forward and backward scroll keys were appropriately labeled.

- User errors and uncertainty about the results of a selection might be reduced if parameter values selected by users (to produce subsequent screens) were displayed in reverse video for a second or two immediately after users designate such a selection through cursor positioning.

4.9.2 LLNL Evaluation

The Vogtle SPDS meets the requirements of Supplement 1 to NUREG-0737 with respect to human factors design.

As discussed in Section 3.3.2 of this TER, GPC should solicit operator feedback concerning the potential areas for improvement listed above. This feedback should be used in deciding if enhancements in these areas are warranted.

5.0 SUMMARY

GPC has installed a SPDS that will provide an extremely effective operator aid. This system completely fulfills most of the SPDS requirements of Supplement 1 to NUREG 0737. Also the verification and validation program has satisfactorily addressed most of the V&V activities recommended by Appendix A to Section 18.2 of NUREG 0800. To allow an unqualified conclusion regarding SPDS acceptability, GPC should, by the first refueling, submit the following items for NRC review.

- o A description of the process and results of feedback obtained from operator experience with SPDS. Design changes as a result of operator feedback concerning the suggested improvement areas listed in Section 3.3.2 should be included in results.

- o A discussion of the final interchannel comparison validity criteria, and a rationale for the choice of each value.

- o An estimate of overall system availability and which considers relevant factors such as maintenance staffing spare parts inventory plans.

- o A description of GPC's review of the V&V programs for the PSMS and PERMS to verify the adequacy of the V&V for the SPDS functions of these systems. Any shortcomings identified by this review and GPC's planned corrective actions should be included in this description.

In addition to the above items LLNL suggests that GPC consider modifying the data validation range checking criteria to reflect the limits of process variation rather than the limits of instrument capability.

6.0 REFERENCES

6.1 GENERAL REFERENCES

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Pertains to Docket Nos. 50-424 and 50-425

13. ABSTRACT (200 words or less)

In June 1985, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-1137) regarding the application of Georgia Power Company, Municipal Electric Authority of Georgia, Oglethorpe Power Corporation, and the City of Dalton, Georgia, for licenses to operate the Vogtle Electric Generating Plant, Units 1 and 2 (Docket Nos. 50-424 and 50-425). Supplement 1 to NUREG-1137 was issued by the staff in October 1985, Supplement 2 was issued in May 1986, Supplement 3 was issued in August 1986, Supplement 4 was issued in December 1986, and Supplement 5 was issued in January 1987. The facility is located in Burke County, Georgia, approximately 26 miles south-southeast of Augusta, Georgia, and on the Savannah River.

This sixth supplement to NUREG-1137 provides recent information regarding resolution of some of the open and confirmatory items that remained unresolved at the time the Safety Evaluation Report was issued.

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