ABB

ASEA BROWN BOVERI

February 18, 1992
LD-92-024

Docket No. 52-002

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Response to NRC Requests for Additional Information

Dear Sirs:

References (A) through (L) requested additional information for the NRC staff review of the Combustion Engineering Standard Safety Analysis Report - Design Certification (CESSAR-DC). Enclosure I to this letter provides our responses to a number of these questions including corresponding revisions to CESSAR-DC. Included in Enclosure I is the response to RAI 430.9 on Electrical Distribution System design. Please note that it may be necessary to revise this response (and the previous responses submitted via letter LD-92-001) when the more-detailed NRC staff position on the second source of offsite power to the non-safety buses is documented. The revised responses may address the specific loads to be supplied by the second offsite power source and the number of turbine generator breakers and unit main transformers.

Should you have any questions on the enclosed material, please contact me or Mr. Stan Ritterbusch of my staff at (203) 285-5206.

Very truly yours,

COMBUSTION ENGINEERING, INC.

C B Brinkman

C. B. Brinkman
Acting Director
Nuclear Systems Licensing

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PDR ADOCK 05200002
A PDR

References:

A) Letter, Plant Systems Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated January 24, 1990

B) Letter, Reactor Systems Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated December 24, 1990

C) Letter, Reactor Systems Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated May 13, 1990

D) Letter, Electrical Systems Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated July 29, 1991

E) Letter, Radiation Protection Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated August 3, 1991

F) Letter, Reactor Safeguards Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated August 5, 1991

G) Letter, Materials and Chemical Engineering Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated August 8, 1991

H) Letter, Instrumentation and Control Systems Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated September 25, 1991

I) Letter, Seismic and Geosciences Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated September 26, 1991

J) Letter, Emergency Preparedness Branch and Standard Plant Project Directorate RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated October 9, 1991

K) Letter, Technical Specifications Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated October 16, 1991

L) Letter, Plant Systems Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated October 10, 1991

vs:
Enclosures: As Stated
cc: J. Trotter (EPRI)
T. Wambach (NRC)

Enclosure I to
LD-92-024

RESPONSE TO NRC REQUESTS FOR ADDITIONAL INFORMATION

Tech Specs.

Question T/S #1

Identify the design differences between the CE System 80+ and the existing CE plants as they relate to the scope of technical specification requirements.

Response T/S #1

The System 80+ design differences related to the scope of the technical specifications are as follows:

1. Safety Injection System

    A. IRWST, CESSAR-DC, Section 6.3.2.2.1

    B. Safety Injection Realignment no longer required, no Recirculation Actuation Signal

    C. Four (4) safety injection pumps instead of 2 LPSI and 2 HPSI.

2. Containment Spray System

    A. Deletion of Iodine Removal System

3. Charging System

    A. Non-Safety System (No Technical Specification needed)

4. Shutdown Cooling System

    A. Independent system, i.e., no use of LPSI pumps for shutdown cooling.

5. Ventilation systems due to plant arrangement are design different and are technical specification affected.

6. Instrumentation and Control

    A. Remote Shutdown Monitoring Instrumentation, CESSAR-DC, Section 7.4.1.1.10.

7. Safety Depressurization System

Question T/S #2

Identify those changes in the technical requirements of the draft STS which CE believes are appropriate or necessary because of the design differences identified under item 1, above. These changes include additional or deleted limiting conditions for operation (LCOs), required remedial actions, completion times for remedial actions, surveillance requirements, and surveillance frequencies.

Response T/S #2

The CESSAR-DC technical specifications provided will be revised in order to take full credit for the design enhancements of the System 80+ arrangement such as:

1. Safety Injection System full flow test capability.
2. Containment Spray/Shutdown Cooling Pump interchange ability.
3. Increased LTOP capacity.
4. Larger Pressurizer Volume.
5. Mid-loop vent valves.
6. Shutdown Cooling System full flow test capability.
7. Emergency Feedwater System full flow test capability
8. Two independent, redundant divisions of CCW, ECW and Emergency Feedwater.
9. Improved Post Accident Monitoring Instrumentation.

The technical specification revisions will be included in a future amendment to CESSAR-DC.

Question 100.2

NUREG-4690 Vol. 1, Rev. 1, Generic Communications Index, presents a list of generic communications from 1971 to 1989 resulting from operating experience at nuclear power plants. Those requiring responses from licensees of operating plants are included as generic letters and bulletins. These issues are incorporated into the SRP, as appropriate, during revisions. There has been no issue of a general revision of the SRP since July 1981; therefore, as part of Chapter 1 of CESSAR-DC, a tabulation of all generic letters and bulletins since January 1981 should be added to address the disposition of these issues for System 80+. If you need a list of generic letters and bulletins for 1990 and 1991, please let me know. The tabulation, similar to what was done in Appendix A for USIs/GSIs, should include categories such as "not applicable" and why, covered under USI/GSI, covered as up-date to a specific section of SRP, covered in CESSAR-DC with specific reference to section or to be covered by interface requirements for COL applicant. In addition, any other improvements made to System 80+ based on operating experience of previously licensed plants should also be listed with appropriate references to the associated CESSAR-DC section.

Response 100.2

Two tables have been prepared to address the Generic Letters and IE Bulletins issued since January 1981. This table identifies the letter or bulletin, its title, and its status with respect to the System 80+ design (i.e., the basis for non-applicability or a reference to a particular section of CESSAR-DC). These tables will be included in a future amendment to Section 1.8 of CESSAR-DC.

A table summarizing improvements in the System 80+ design which were based on operating experience is attached. Included is the collective experience of the industry as promulgated through the EPRI Utility Requirements Document as well as designer-specific experience. This table will be included in Section 1.2 of CESSAR-DC in a future amendment.

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and indicates those that

1.8 REGULATORY GUIDES

identifies all U.S. NRC

Table 1.8-1 ~~lists the applicable~~ Regulatory Guides, ~~which~~ are ~~addressed in~~ the System 80+ Standard Design. B E

applicable to

Insert Y

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Insert Y

Tables 1.8-2 and 1.8-3 document the applicability of Generic Letters and IE Bulletins to the System 80+ Standard Design. An item was considered "not applicable" if it met one or more of the following criteria.

a. The item specifically identifies another design or vendor.

b. The item is specific to components, structures or systems which are not included in the System 80+ Standard Design.

c. The item is relevant to plant operations or is specific to a particular plant design.

d. The item includes no design requirements.

e. The item is superseded by another item.

f. The item is not mandatory but is an alternative which can be implemented as desired by an applicant.

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Table 1.8-2

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+

No.	Title	Comment
91-19	Information to Addressees Regarding New Telephone Numbers for NRC Offices Located in One White Flint North	N/A (d)
91-18	Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability	N/A (d)
91-17	Generic Safety Issue 29, "Bolting Degradation or Failures in Nuclear Power Plants"	Appendix A (GI-29)
91-16	Licensed Operators' and Other Nuclear Facility Personnel Fitness for Duty	N/A (c)
91-15	Operating Experience Feedback Report, Solenoid-Operated Valve Problems at U.S. Reactors	N/A (d)
91-14	Emergency Telecommunications	N/A (d)
91-13	Request for Information Related to the Resolution of Generic Issue 130, "Essential Service Water System Failures at Multi-Unit Sites," Pursuant to 10CFR50.54(f)	N/A (d)
91-12	Operator Licensing National Examination Schedule	N/A (c)
91-11	Resolution of Generic Issues 48, "LCOs for Class 1E Vital Instrument Buses," and 49, "Interlocks and LCOs for Class 1E Tie Breakers" Pursuant to 10 CFR 50.54(f)	Appendix A (GI-48, GI-49)
91-10	Explosives Searches at Protected Area Portals	N/A (c)

Table 1.8-2

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
91-09	Modification of Surveillance Interval for the Electrical Protective Assemblies in Power Supplies for the Reactor Protection System	N/A (a)
91-08	Removal of Component Lists from Technical Specifications	16.0
91-07	GI-23, "Reactor Coolant Pump Seal Failures" and Its Possible Effect on Station Blackout	Appendix A (GI-23, A-44)
91-06	Resolution of Generic Issue A-30, "Adequacy of Safety-Related DC Power Supplies" Pursuant to 10 CFR 50.54(f)	Appendix A (A-30), 8.3.2
91-05	License Commercial - Grade Procurement and Dedication Programs	N/A (d)
91-04	Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle	N/A, 18 Month Fuel Cycle (f)
91-03	Reporting of Safeguards Events	N/A (d)
91-02	Reporting Mishaps Involving LLW Forms Prepared for Disposal	N/A (c)
91-01	Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications	10.7
90-09	Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions	N/A, Inspection Intervals based on 18 Month refueling cycle (f)
90-08	Simulation Facility Exemptions	N/A (c)
90-07	Operator Licensing National Examination Schedule	N/A (c)

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
90-06	Resolution of Generic Issue 70, "Power Operated Relief Valve and Block Valve Reliability", and Generic Issue 94, "Additional Low - Temperature Overpressure Protection for Light - Water Reactors", Pursuant to 10 CFR 50.54(f)	5.4.13, 5.2.2.10, Appendix A (GI-094)
90-05	Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping	N/A (c)
90-04	Request for Information on the Status of License Implementation of Generic Safety Issues Resolved with Imposition of Requirements or Corrective Actions	Applicable GIs in Appendix A
90-03 Supp. 1	Relaxation of Staff Position in Generic Letter 83-28, Item 2.2 Part 2 "Vendor Interface for Safety-Related Components"	17.0
90-03	Relaxation of Staff Position in Generic Letter 83-28, Item 2.2 Part 2 "Vendor Interface for Safety-Related Components"	17.0
90-02	Alternative Requirements for Fuel Assemblies in the Design Features Section of Technical Specifications	N/A (f)
90-01	Request for Voluntary Participation in the NRC Regulatory Impact Survey	N/A (d)
89-23	NRC Staff Response to Questions Pertaining to Implementation of 10 CFR Part 26	N/A (d)

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
89-22	Potential for Increased Roof Loads and Plant Area Flood Runoff Depth at Licensed Nuclear Power Plants Due to Recent Change in Probable Maximum Precipitation Criteria Developed by the National Weather Service	2.4
89-21	Request for Information Concerning Status of Implementation of Unresolved Safety Issue Requirements	Applicable USIs in Appendix A
89-20	Protected Area Long-Term Housekeeping	N/A (c)
89-19	Request for Action Related to Resolution of Unresolved Safety Issue A-47 "Safety Implication of Control System in LWR Nuclear Power Plant." Pursuant to 10 CFR 50.54(f)	Appendix A (A-47)
89-18	Resolution of Unresolved Safety Issue A-17, "Systems Interactions in Nuclear Power Plants".	Appendix A (A-17)
89-17	Planned Administrative Changes to the NRC Operator Licensing Written Examination Process	N/A (c)
89-16	Installation of a Hardened Wetwell vent	N/A (a)
89-15	Emergency Response Data System	N/A (c)
89-14	Line-Item Improvements in Technical Specifications - Removal of the 3.25 Limit on Extending Surveillance Intervals	16.3.2 (SR3.0.2)
89-13	Service Water System Problems Affecting Safety-Related Equipment	9.2.1
89-12	Operator Licensing Examinations	N/A (c)

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
89-11	Resolution of Generic Issue 101 "Boiling Water Reactor Water Level Redundancy"	N/A (a)
89-10	Safety-Related Motor-Operated Valve Testing and Surveillance	3.9.3.2
89-09	ASME Section III Component Replacements	N/A (c)
89-08	Erosion/Corrosion - Induced Pipe Wall Thinning	3.6
89-07	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	Appendix 13A
89-06	Task Action Plan Item I.D.2 - Safety Parameter Display System - 10 CFR 50.54(f)	Appendix A (I.D.2)
89-05	Pilot Testing of the Fundamentals Examination	N/A (c)
89-04	Guidance on Developing Acceptable Inservice Testing Programs	6.6
89-03	Operator Licensing National Examination Schedule	N/A (c)
89-02	Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products	N/A (c)
89-01	Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program	N/A (c)

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
88-20 Supp. 4	Individual Plant Examination of External Events for Severe Accident Vulnerabilities	Appendix B Section 4.0
88-20 Supp. 3	Completion of Containment Performance Improvement Program and Forwarding of Insights for use in the Individual Examination for Severe Accident Vulnerabilities	Appendix B Section 5.0
88-20 Supp. 2	Accident Management Strategies for Consideration in the Individual Plant Examination Process	Provides input to the Severe Accident Management Guide.
88-20 Supp. 1	Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)	N/A, Issue covered by PRA (Appendix B)
88-20	Individual Plant Examination for Severe Accident Vulnerabilities	N/A, Issue covered by PRA (Appendix B)
88-19	Use of Deadly Force by License Guards to Prevent Theft of SNM.	N/A (c)
88-18	Plant Record Storage on Optical Discs	N/A (c)
88-17	Loss of Decay Heat Removal	5.4.7, 7.7.1
88-16	Removal of Cycle-Specific Parameter Limits from Technical Specifications	16.0
88-15	Electric Power Systems-Inadequate Control over Design Process	8.0 and PRA design process.
88-14	Instrument Air Supply System Problems Affecting Safety-Related Equipment	9.3.1
88-13	Operator Licensing Examination	N/A (c)
88-12	Removal of Fire Protection Requirements from Technical Specifications	16.0

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
88-11	NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations	5.3.1.6
88-10	Purchare of GSA Approved Security Containers	N/A (c)
88-09	Pilot Testing of Fundamentals Examinations	N/A (c)
88-08	Mail Sent or Delivered to the Office of Nuclear Reactor Regulation	N/A (d)
88-07	dified Enforcement Policy lating to 10 CFR 50.49, nvironmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants"	N/A (d)
88-06	Removal of Organization Charts from Technical Specification Administrative Control Requirements	N/A (f)
88-05	Boric Acid Corrosion of Carbon Steel Reactor Vessel Boundary Components in PWR Plants	3.11
88-04	Distribution of Gems Irradiated in Research Reactors	N/A (a)
88-03	Resolution of Generic Safety Issue 93, "Steam Bindings of Auxiliary Feedwater Pumps"	Appendix A (GI 93)
88-02	Integrated Safety Assessment Program II	Level III PRA, Appendix B
88-01	NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping	N/A (a)

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
87-16	Transmittal of NUREG-1262, "Answers to Questions at Public Meeting Regarding Implementation of Title 10, Code of Federal Regulations, Part 55 on Operators' Licenses"	N/A (d)
87-15	Policy Statement on Deferred Plants	N/A (c)
87-14	Operator Licensing Examinations	N/A (c)
87-13	Integrity of Requalification Examinations at Non-Power Reactors	N/A (c)
87-12	Loss of Residual Removal While the Reactor Coolant System is Partially Filled.	N/A (e) Superseded by G.L. 88-17
87-11	Relaxation in Arbitrary Intermediate Pipe Rupture Requirements	3.6.2, 3.6.3
87-10	Implementation of 10 CFR 73.57, Requirements for FBI Criminal History Checks	N/A (c)
87-09	Sections 3.0 and 4.0 of the Standard Technical Specifications on the Applicability of Limiting Conditions for Operation and Surveillance Requirements	16.3
87-08	Implementation of 10 CFR 73.55 Miscellaneous Amendments and Search Requirements	N/A (c)
87-07	Information Transmittal of Final Rulemaking for Revisions to Operator Licensing - 10 CFR 55 and Conforming Amendments	N/A (c)
87-06	Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves	N/A (c)

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
87-05	Request for Additional Information - Assessment of License Measures to Mitigate and/or Identify Potential Degradation of MARK I Drywells	N/A (a)
87-04	Temporary Exemption from Provisions of the FBI Criminal History Rule for Temporary Workers	N/A (c)
87-03	Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue A-46	N/A (d)
87-02	Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors	N/A (d)
87-01	Public Availability of the NRC Operator Licensing Examination Question Bank	N/A (c)
86-17	Availability of NUREG-1169	N/A (a)
86-16	Westinghouse ECC Evaluation Models	N/A (a)
86-15	Information Relating to Compliance with 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants".	N/A (e), Superseded by G. L. 88-07
86-14	Operator Licensing Examinations	N/A (c)
86-13	Potential Inconsistency between Safety Analyses and Technical Specifications	N/A (d)
86-12	Criteria for Unique Purpose Exemption from Conversion from the Use of HEU Fuel	N/A (d)
86-11	Distribution of Products Irradiated in Research Reactors	N/A (a)

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
86-10	Implementation of Fire Protection Requirements	9.5.1
86-09	Technical Resolution of Generic Issue No. B-59-(N-1) Loop Operation in BWRs and PWRs	N/A (f)
86-08	Availability of Supplement 4 to NUREG-0933, "A Prioritization of Generic Safety Issues"	Applicable GIs in Appendix A
86-07	Transmittal of NUREG-1160 Regarding the San Onofre Unit 1 Loss of Power and Water Hammer Event	N/A (d)
86-06	Implementation of TMI Action Item II.K.3.5, "Automatic Trip to Reactor Coolant Pumps"	Appendix A (II.K.3)
86-05	Implementation of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" [B&W]	N/A (a)
86-04	Policy Statement on Engineering Expertise on Shift	N/A (c)
86-03	Applications for License Amendments	N/A (c)
86-02	Technical Resolution of Generic Issue B-19 - Thermal Hydraulic Stability [BWR]	N/A (a)
86-01	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	N/A (a)
85-22	Potential for Loss of Post-LOCA Recirculation Capability due to Insulation Debris Blockage	N/A (d)
85-21	Not Issued.	
85-20	Resolution of Generic Issue 69 [B&W]	N/A (a)
85-19	Reporting Requirements on Primary Coolant Iodine Spikes	16.3.4.15

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
85-18	Operator Licensing Examinations	N/A (c)
85-17	Availability of Supplement 2 and 3 to NUREG-0933	N/A (e), Superceded by G. L. 86-08
85-16	High Boron Concentrations	N/A (c)
85-15	Information Relating to the Deadlines for Compliance with 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plant"	N/A (e), Superseded by G. L. 88-07
85-14	Commercial Storage at Power Reactor Sites of Low-Level Radioactive Waste Not Generated by the Utility	N/A (c)
85-13	Transmittal of NUREG-1154 Regarding the Davis-Besse Loss of Main and Auxiliary Feedwater Event	N/A (d)
85-12	Implementation of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" [Westinghouse]	N/A (a)
85-11	Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants" NUREG-0612	N/A (d)
85-10	Technical Specifications for Generic Letter 83-28, Items 4.3 and 4.4 [B&W]	N/A (a)
85-09	Revision of NRC Form 439, "Report of Terminating Individual's Occupational Exposure"	N/A (c)
85-08	10 CFR 20.408 Termination Reports - Format	N/A (c)
85-07	Implementation of Integrated Schedules for Plant Modifications	N/A (c)

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
85-06	Quality Assurance Guidance for ATWS Equipment that is not Safety-Related	N/A (d)
85-05	Inadvertent Boron Dilution Events	15.4.6
85-04	Operator Licensing Examinations	N/A (c)
85-03	Clarification of Equivalent Control Capacity for Standby Liquid Control Systems [BWR]	N/A (a)
85-02	Staff Recommended Actions Stemming from NRC Integrated program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity	Appendix A (A-4), 5.4.2, 7.7.1.6, 10.3.5
85-01	Fire Protection Policy Steering Committee Report	9.5.1
84-24	Certification of Compliance to 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"	N/A (c)
84-23	Reactor Vessel Water Level Instrumentation in BWRs	N/A (a)
84-22	Not Issued.	
84-21	Long Term Low Power Operation in Pressurized Water Reactors	16.0
84-20	Scheduling Guidance for License Submittals of Reloads that Involve Unreviewed Safety Questions	N/A (d)
84-19	Availability of Supplement 1 to NUREG-0933, "A Prioritization of Generic Safety Issues"	Appendix A
84-18	Filing of Applications for Licenses and Amendments	N/A (d)

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(Sheet 13 of 22)

GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
84-17	Annual Meeting to Discuss Recent Development Regarding Operator Training, Qualifications, and Examinations	N/A (c)
84-16	Adequacy of On-Shift Operating Experience for Near Term Operating License Applicants	N/A (c)
84-15	Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability	8.3.1.1
84-14	Replacement and Requalification Training Program	N/A (c)
84-13	Technical Specification for Snubbers	Appendix A (A-13)
84-12	Compliance with 10 CFR Part 61 and Implementation of the Radiological Effluent Technical Specifications and Attendant Process Control Program	N/A (c)
84-11	Inspections of BWR Stainless Steel Piping	N/A (a)
84-10	Administration of Operating Tests Prior to Initial Criticality	N/A (c)
84-09	Recombiner Capability Requirements of 10 CFR 50.44(c)(3)(II)	6.2.5
84-08	Interim Procedures for NRC Management of Plant - Specific Backfitting	N/A (d)
84-07	Procedural Guidance for Pipe Replacement at BWRs	N/A (a)
84-06	Operator and Senior Operator License Examination Criteria for Passing Grade	N/A (c)
84-05	Change to NUREG-1021, "Operator Licensing Examiner Standards"	N/A (c)

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
84-04	Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops	N/A (f)
84-03	Availability of NUREG-0933, "A Prioritization of Generic Safety Issues"	N/A (e), Superceded by G. L. 85-17
84-02	Notice of Meeting Regarding Facility Staffing	N/A (c)
84-01	NRC Use of the Terms, "Important to Safety and Safety Related"	N/A (d)
83-44	Availability of NUREG-1021, "Operator Licensing Examiner Standards"	N/A (c)
83-43	Reporting Requirements of 10 CFR Part 50, Sections 50.72 and 50.73, and Standard Technical Specifications	N/A (d)
83-42	Clarification to Generic Letter 81-07 Regarding Response to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"	9.1.4
83-41	Fast Cold Starts of Diesel Generators	N/A (d)
83-40	Operator Licensing Examination	N/A (c)
83-39	Voluntary Survey of Licensed Operators	N/A (c)
83-38	NUREG-0965, "NRC Inventory of Dams"	N/A (d)
83-37	NUREG-0737 Technical Specifications	16.0
83-36	NUREG-0737 Technical Specifications [BWR]	N/A (a)
83-35	Clarification of TMI Action Plan Item II.K.3.31	N/A (d)

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
83-34	Not Issued.	
83-33	NRC Positions on Certain Requirements of Appendix R to 10 CFR 50	9.5.1
83-32	NRC Staff Recommendations Regarding Operator Action for Reactor Trip and ATWS	9.9.1.1.11
83-31	Safety Evaluation of 'Abnormal Transient Operating Guidelines" [B&W]	N/A (a)
83-30	Deletion of Standard Technical Specification Surveillance Requirement 4.8.1.1.2.d.6 of Diesel Generator Testing	N/A (f)
83-29	Not Issued.	
83-28	Required Actions Based on Generic Implications of Salem ATWS Events	N/A (c)
83-27	Surveillance Intervals in Standard Technical Specifications	N/A (e), Superseded by G. L. 91-04
83-26	Clarification of Surveillance Requirements for Diesel Fuel Impurity Level Tests	N/A (d)
83-25	Not Issued.	
83-24	TMI Task Action Item I.G.1, "Special Low Power Testing and Training", Recommendations for BWRs	N/A (a)
83-23	Safety Evaluation of "Emergency Response Guidelines"	N/S (c)
83-22	Safety Evaluation of "Emergency Response Guidelines" [Westinghouse]	N/A (a)
83-21	Clarification of Access Control Procedures for Law Enforcement Visits	N/A (d)

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Table 1.8-2

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
83-20	Integrated Scheduling for Implementation of Plant Modifications	N/A (d)
83-19	New Procedures for Providing Public Notice Concerning Issuance of Amendment to Operating Licenses	N/A (d)
83-18	NRC Staff Review of the BWR Owner's Group Control Room Survey Program	N/A (a)
83-17	Integrity of the Requalification Examination for Removal of Reactor Operator and Senior Reactor Operator Licenses	N/A (c)
83-16	Transmittal of NUREG-0977 Relative to the ATWS Events at Salem Generating Station, Unit No. 1	N/A (d)
83-15	Implementation of Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds during Preservice and Inservice Examinations, Revision 1"	5.1.4, 5.3.1
83-14	Definition of "Key Maintenance Personnel"	N/A (c)
83-13	Clarification of Surveillance Requirements for HEPA Filters and Charcoal Absorber Units in Standard Technical Specifications on GSF Cleanup Systems	16.10.14, 16.10.15
83-12A	Issuance of NRC Form 398 - Personal Qualifications Statement - License	N/A (d)
83-12	Issuance of NRC Form 398 - Personal Qualifications Statement - License	N/A (d)

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Table 1.8-2

(Sheet 17 of 22)

GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
83-11	License Qualification for Performing Safety Analyses in Support of Licensing Actions	N/A (d)
83-10b	Resolution of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps"	N/A (e), Superseded by G. L. 86-06
83-10a	Resolution of TMI Action Item II.K.3.5., "Automatic Trip of Reactor Coolant Pumps"	N/A (e), Superseded by G. L. 86-06
83-09	Review of Combustion Engineering Owner's Group Emergency Procedures Guideline Program	N/A (e), Superseded by G. L. 83-23
83-08	Modification of Vacuum Breakers on Mark I Containments	N/A (a)
83-07	The Nuclear Waste Policy Act of 1982	N/A (c)
83-06	Certificates and Revised Format for Reactor Operator and Senior Reactor Operator Licenses	N/A (c)
83-05	Safety Evaluation of "Emergency Procedure Guidelines, Revision 2", NEDO-24934, June 1982 [BWRs]	N/A (a)
83-04	Regional Workshops Regarding Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability	N/A (d)
83-03	Not Issued.	
83-02	NUREG-0737 Technical Specifications [BWRs]	N/A (a)
83-01	Operator Licensing Examination Site Visit	N/A (c)
82-39	Problems with the Submittals of 10 CFR 72.21 Safeguards Information for Licensing Review	N/A (c)
82-38	Meeting to Discuss Recent Developments for Operating Licensing Examinations	N/A (c)

Table 1.8-2

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
82-37	Not Issued.	
82-36	Not Issued.	
82-35	Not Issued.	
82-34	Not Issued.	
82-33	Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability	Appendix A
82-32	Subjects Requirements	N/A (d)
82-31	Not Issued.	
82-30	Filing Relating to 10 CFR 50 Production and Utilization Facilities	N/A (d)
82-29	Not Issued.	
82-28	Inadequate Core Cooling Instrumentation System	6.3.5, 7.5
82-27	Transmittal of NUREG-0763 and NUREG-0783 [BWRs]	N/A (a)
82-2[illegible]	NUREG-0744 Rev. 1 - Pressure Vessel Material Fracture Toughness	5.3.1
82-25	Integrated IAEA Exercise for Physical Inventory at LWRs	N/A (d)
82-24	Safety/Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments	N/A (a)
82-23	Inconsistency between Requirements of 10 CFR 73.40(d) and Standard Technical Specifications for Performing Audits for Safeguards Contingency plans	N/A (d)
82-22	Steam Generator Tube Integrity	N/A (c)
82-21	Technical Specifications for Fire Protection Audits	N/A (c)

Table 1.8-2

(Sheet 19 of 22)

GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
82-20	Guidance for Implementing Standard Review Plan Rule	N/A (d)
82-19	Submittal of Copies of Document to NRC	N/A (d)
82-18	Reactor Operator and Senior Reactor Operator Requalification Examinations	N/A (c)
82-17	Inconsistency between Requirements of 10 CFR 50.54(t) and Standard Technical Specifications for Performing Audits of Emergency Preparedness Programs	N/A (c)
82-16	NUREG-0737 Technical Specifications	N/A (d)
82-15	Not Issued.	
82-14	Submittal of Documents to the Nuclear Regulatory Commission	N/A (d)
82-13	Reactor Operator and Senior Reactor Operator Examinations	N/A (c)
82-12	Nuclear Power Plant Staff Working Hours	N/A (c)
82-11	Transmittal of NUREG-0916 Relative to the Restart of R. E. Ginna Nuclear Power Plant	N/A (d)
82-10	Post-TMI Requirements	Appendix A
82-09	Environmental Qualification of Safety-Related Electrical Equipment	N/A (d)
82-08	Transmittal of NUREG-0909 Relative to the Ginna Tube Rupture	N/A (d)
82-07	Transmittal of NUREG-0909 Relative to the Ginna Tube Rupture	N/A (d)
82-06	Not Issued.	

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Table 1.8-2

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
82-05	Post-TMI Requirements	N/A (e), Superseded by G. L. 82-10
82-04	Use of SEE-IN Program	N/A (d)
82-03	High Burnup MAPLHGR Limits [BWRs]	N/A (a)
82-02	Nuclear Power Plant Staff Working Hours	N/A (c)
82-01	New Applications Survey	N/A (d)
81-40	Qualifications of Reactor Operators	N/A (c)
81-39	NRC Volume Reduction Policy	N/A (d)
81-38	Storage of Low Level Radioactive Wastes at Power Reactor Site	N/A (c)
81-37	ODYN Code Reanalysis Requirements [BWRs]	N/A (a)
81-36	Revised Schedule for Completion of TMI Action Plan Item II.D.1, Relief and Safety Valve Testing	5.4.13
81-35	Safety Concerns Associated with Pipe Breaks in the BWR Scram System.	N/A (a)
81-34	Safety Concerns Associated with Pipe Breaks in the BWR Scram System.	N/A (a)
81-33	Not Issued.	
81-32	NUREG-0737, Item II.K.3.44 - Evaluation of Anticipated Transients Combined with Single Failure [BWRs]	N/A (a)
81-31	Not Issued.	
81-30	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	N/A (a)
81-29	Simulator Examinations	N/A (c)

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Table 1.8-2

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
81-28	Steam Generator Overfill	N/A (c)
81-27	Privacy and Proprietary Material in Emergency Plans	N/A (c)
81-26	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	N/A (a)
81-25	Change in Implementing Schedule for Submission and Evaluation of Upgraded Emergency plans	N/A (c)
81-24	Multi-Plant Issue B-56 Control Rods Fail to Fully Insert [BWRs]	N/A (a)
81-23A	INPO Evaluation Reports	N/A (d)
81-23	INPO Plant Specifi Evaluation Report	N/A (d)
81-22	Engineering Evaluation of the H. B. Robinson Reactor Coolant System Leak on 1/27/81	N/A (d)
81-21	Natural Circulation Cooldown	5.4.1
81-20	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	N/A (a)
[illegible]-19	Thermal Shock to Reactor Pressure Vessels	5.3.1
81-18	BWR Scram Discharge System - Clarification of Diverse Instrumentation Requirements	N/A (a)
81-17	Functional Criteria for Emergency Response Facilities	N/A (c)
81-16	NUREG-0737 Item I.C.1 SER on Abnormal Transient Operating Guidelines [B&W]	N/A (a)
81-15	Environmental Qualification of Class 1E Electrical Equipment - Clarification of Staffs Handling of Proprietary Information	N/A (d)

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Table 1.8-2

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GENERIC LETTERS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
81-14	Seismic Qualification for Auxiliary Feedwater System	10.4.9
81-13	SER for GEXL Correlation for 8x8R Fuel Reload Applications	N/A (a)
81-12	Fire Protection Rule	N/A (d)
81-11	BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking	N/A (a)
81-10	Post-TMI Requirements for the Emergency Operations Facility	N/A (c)
81-09	BWR Scram Discharge System	N/A (a)
81-08	ODYN Code [BWRs]	N/A (a)
81-07	Control of Heavy Loads	9.1.4
81-06	Periodic Updating of Final Safety Analysis Reports	N/A (c)
81-05	Information Regarding the Program for Environmental Qualification of Safety-Related Electrical Equipment	N/A (c)
81-04	Emergency Procedures and Training for Station Blackout Events	N/A (c)
81-03	Implementation of NUREG-0313 [BWRs]	N/A (a)
81-02	Analysis, Conclusion and Recommendation Concerning Operator Licensing	N/A (c)
81-01	Qualification of Inspection, Examination and Audit Personnel	N/A (c)

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Table 1.8-3

(Sheet 1 of 4)

IE BULLETINS APPLICABILITY ANALYSIS TO SYSTEM 80+

No.	Title	Comment
91-01	Reporting Loss of Criticality Safety Controls	N/A (a)
90-02	Loss of Thermal Margin Caused by Channel Box Bow [BWRs]	N/A (a)
90-01	Loss of Fill-Oil in Transmitters Manufactured by Rosemount	N/A (b)
89-03	Potential Loss of Required Shutdown Margin during Refueling Operation	N/A (c)
89-02	Stress Corrosion Cracking of High - Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves of Similar Design	N/A (c)
89-01	Failure of Westinghouse Steam Generator Tube Mechanical Plugs	N/A (a)
88-11	Pressurizer Surge Line Thermal Stratification	5.2, 5.4
88-10 and Suppl.	Nonconforming Molded - Case Circuit Breakers	N/A (c)
88-09	Thimble Tube Thinning in Westinghouse Reactors	N/A (a)
88-08 and Suppl.	Thermal Stress in Piping Connected to Reactor Coolant Systems	3.9
88-07 and Suppl.	Power Oscillations in Boiling Water Reactors	N/A (a)
88-06	Actions to be taken for the Transportation of Model No. Spec 2-T Radio Graphic Exposure Device	N/A (b)
88-05 and Suppl.	Nonconforming Materials Supplied by Piping Suppliers, Inc.	N/A (b)

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Table 1.8-3

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IE BULLETINS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
88-04	Potential Safety - Related Pump Loss [Westinghouse]	RAI 440.71
88-03	Inadequate Latch Engagement in HFA Type Latching Relays Manufactured by GE Company	N/A (a)
88-02	Rapidly Propagating Fatigue Cracks in Steam Generator Tubes [Westinghouse]	N/A (a)
88-01	Defects in Westinghouse Circuit Breakers	N/A (a)
87-02 and Suppl.	Fastener Testing to Determine Conformance with Applicable Material Specifications	N/A (c)
87-01	Thinning of Pipe Walls in Nuclear Power Plants	[illegible].6
86-04	Defective Teletherapy Timer that May Not Terminate Treatment Dose	N/A (b)
86-03	Potential Failure of Multiple ECCS Pumps due to Single Failure of Air - Operated Valve in Minimum Flow Recirculation Line	N/A (d)
86-02	State "O" Ring Differential Pressure Switches	N/A (b)
86-01	Minimum Flow Logic Problems that Could Disable RWR Pumps [BWRs]	N/A (a)
85-03 and Suppl.	Motor - Operated Valve Common Failures during Plant Transients due to Improper Switch Settings	N/A (c)
85-02	Undervoltage Trip Attachments of Westinghouse DB-50 Type Reactor Trip Breakers	N/A (a)
85-01	Steam Binding of Auxiliary Feedwater Pumps	10.4.9
84-03	Refueling Cavity Water Seal	9.1
84-02	Failures of General Electric Type HFA Relays in Use in Class 1E Safety Systems	N/A (a)

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Table 1.8-3

(Sheet 3 of 4)

IE BULLETINS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
84-01	Cracks in Boiling Water Reactor Mark I Containment Vent Headers	N/A (a)
83-08	Electrical Circuit Breakers with an Undervoltage Trip Feature	N/A (c)
83-07 and Suppl.	Apparently Fraudulent Products Sold by Ray Miller, Inc.	N/A (b)
83-06	Nonconforming Materials Supplied by Tube-Line Corporation Facilities	N/A (b)
83-05	ASME Code Pumps and Spare Parts Manufactured by the Haywood Tyler Pump Company	N/A (b)
83-04	Failure of Undervoltage Trip Function of Reactor Trip Breakers	N/A (c)
83-03	Check Valve Failures in Raw Water Cooling Systems of Diesel Generators	9.5.5
83-02	Stress Corrosion Cracking in Large - Diameter Stainless Steel Recirculation System Piping at BWR Plants	N/A (a)
83-01	Failure of Trip Breakers to Open on Automatic Trip Signal	N/A (c)
82-04	Deficiencies in Primary Containment Electrical Penetration Assemblies	N/A (c)
82-03	Stress Corrosion Cracking in Thick - Wall, Large Diameter, Stainless Steel, Recirculation System Piping at BWR Plants	N/A (a)
82-02	Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants	N/A (c)
82-01 and Rev.	Alteration of Radiographs of Welds in Piping Subassemblies	N/A (c)
81-03	Flow Blockage of Cooling Water to Safety System Components	N/A (c)

Table 1.8-3

(Sheet 4 of 4)

IE BULLETINS APPLICABILITY ANALYSIS TO SYSTEM 80+ (Cont'd)

No.	Title	Comment
81-02 and Suppl.	Failure of Gate Type Valves to Close against Differential Pressure	N/A (c)
81-01	Surveillance of Mechanical Snubbers	N/A (c)

and in Table 1.2-1.

1.2.2 SYSTEM 80+ STANDARD DESIGN - SCOPE AND DESCRIPTION

The design scope of the System 80+ Standard Design includes all buildings, structures, systems, and components which can significantly affect safe operation. The primary design characteristics are summarized in the subsections below. The seismic category, safety classification, and quality classification of mechanical components are listed in Table 3.2-1. E

1.2.3 NUCLEAR STEAM SUPPLY SYSTEM (NSSS)

The NSSS generates approximately 3817 Mwt, producing saturated steam.

The NSSS contains two independent primary coolant loops, each of which has two reactor coolant pumps, a steam generator, a 42-inch ID outlet (hot) pipe and two 30-inch ID inlet (cold) pipes. In addition, the safety injection lines are connected directly to the Reactor Vessel. An electrically heated pressurizer is connected to one of the loops of the NSSS. The pressurizer has an increased volume to enhance transient response. Pressurized water is circulated by means of electric-motor-driven, single-stage, centrifugal reactor coolant pumps. Reactor coolant flows downward between the reactor vessel shell and the core support barrel, upward through the reactor core, through the hot leg piping, through the tube side of the vertical U-tube steam generators, and back to the reactor coolant pumps. The saturated steam produced in the steam generators is passed to the turbine. A E

1.2.3.1 Reactor Core

The reactor core is fueled with uranium dioxide pellets enclosed in zircaloy tubes with welded end caps. The tubes are fabricated into assemblies in which end fittings limit axial motion and grids limit lateral motion of the tubes. The control element assemblies (CEAs) consist of NiCrFe alloy clad boron carbide absorber rods, or hafnium full strength absorber rods and solid NiCrFe alloy reduced strength absorber rods, which are guided by tubes located within the fuel assembly. The core consists of 241 fuel assemblies which will be initially loaded with three different U-235 enrichments. The NSSS full thermal output is 3817 MWt with a core thermal output of 3800 MWt. A

Design criteria are established to ensure the following:

A. The minimum departure from nucleate boiling ratio during normal operation and anticipated operational occurrences will provide at least a 95% probability with 95% confidence that departure from nucleate boiling does not occur. B

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Table 1.2-1

System 80+ Improvements Based on Operating Experience

This table summarizes those design improvements which have resulted from design and analysis experience as well as plant startup and operating experience. This experience reflects both industry experience via the EPRI Utility Requirements Document and designer-specific experience.

1. Integrated Design Process:

   - One organization, Combustion Engineering, Inc., is responsible for the design of structures, systems, and components of a plant which are important to safety (where design features depend on site-specific characteristics, interface requirements are provided), thus facilitating an integrated design process. The major considerations in this integrated design approach are as follows:

     - The PRA is used to evaluate the design and to identify areas where significant improvement can be obtained. Although the end product of the PRA is a calculation of core damage frequency and offsite consequences, the PRA can also be used to gain design insights and identify improvements for handling more frequent transients and accidents (Appendix B).

     - Maintainability of the plant is being addressed by using equipment that minimizes the need for maintenance, by assuring that equipment can be easily accessed, and by assuring that maintenance actions will be as simple as possible (so as to avoid unplanned reactor trips and plant downtime). These same considerations apply to periodic testing and inspection of equipment.

     - In almost all cases for System 80+, safety and non-safety functions have been separated. This will make the plant much simpler to operate and maintain.

     - Human factors (i.e., the man-machine interface) are considered throughout the plant and especially in the control room (Chapter 18).

     - ALARA consideration affect the selection of materials and location of piping and equipment that carry radioactive coolant. For example, specifications for the reactor coolant system materials have been tightened to minimize transport of contamination. Improvements in the steam generator tubing material and access openings will greatly reduce radiation

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- exposures for maintenance, testing, and inspection. The overall goal is to maintain personnel exposure to less than 100 man-rems per year for each reactor (Chapter 12).

- Plant security (i.e., sabotage protection) and fire protection concerns have been directly addressed in determining layouts for plant safety systems (Section 13.6).

2. Increased RCS Design Margins and Improvements:

- Reactor: The core operating margin has been increased by reducing the normal operating hot leg temperature and revising core parameter monitoring methods. The ability to change operating power level (i.e., maneuver) using control rods only (without adjusting boron concentration in the coolant system) has been provided, simplifying reactivity control during plant load changes and reducing liquid waste processing requirements (Sections 4.3 and 4.4).

- Reactor Pressure Vessel: The reactor vessel is ring-forged with material specifications that result in a sixty year end-of-life RT_{NDT} well below the current NRC screening criteria. This results in a significant reduction in the number of welds (with resulting reduction in inservice inspection) and eliminates concern for pressurized thermal shock (Section 5.3).

- Pressurizer: The pressurizer volume is increased to enhance transient response and reduce unnecessary challenges to safety systems (Section 5.4.10).

- Steam Generators: The steam generators include Inconel 690 tubes, improved steam dryers, and a seventeen percent increase in overall heat transfer area, including a ten percent margin for potential tube plugging. The steam generators have a twenty-five percent larger secondary feedwater inventory to extend the "boil dry" time and improve response to upset conditions. Steam generator improvements also have been added to facilitate maintenance and long term integrity. These include larger and repositioned manways, a standby recirculation nozzle, and a redesigned flow distribution plate (Section 5.4.2).

- Mechanical improvements based on System 80 startup and operating experience include strengthened reactor coolant pump impellers, redesigned reactor coolant temperature detector thermowells, strengthened reactor vessel upper guide structure, specification of antimony-free reactor coolant pump bearings, strengthened reactor coolant pump shafts, and redesigned steam generator economizer internals.

3. <u>Advanced Control Room Design:</u>

- The Advanced Control Complex (Nuplex 80+) for System 80+ has been designed to meet demanding human factor, reliability, and licensing requirements, and is characterized by state-of-the-art advances, such as distributed digital processing, fiber optic data communications, and touch sensitive video displays (Chapter 18).

- Nuplex 80+ is a total integration of plant-wide instrumentation and controls (I&C) systems. The Advanced Control Complex includes the Main Control Room, the Technical Support Center, the Remote Shutdown Control Room, and the I&C Equipment Rooms (which contain control, protection, and monitoring systems).

- Redundancy and diversity in all information processing and display ensures the correctness of information presentation and allows continued operation with equipment failures. The integration of information from the former Safety Parameter Display System and the Post Accident Monitoring Instrumentation (PAMI) into normal operating displays allows the same displays to be used during all plant conditions.

- Alarms are based on validated signal inputs with logic and setpoints that account for plant and equipment operating modes. Four levels of alarm presentation are employed. Individual and global alarm acknowledgement features ensure that all alarms are recognized without operator task overload. Alarm acknowledgement provides direct access to supporting displays.

4. <u>Highly Reliable Safeguards Systems:</u>

- Chemical and Volume Control System (CVCS): The CVCS incorporates numerous significant improvements which include centrifugal charging pumps, a high pressure letdown heat exchanger, and simplified charging and auxiliary spray piping. Required safety functions previously performed by the CVCS are now delegated to other dedicated safety systems (Section 9.3.4).

- Safety Injection System (SIS): The SIS design has been improved to provide a simpler and more reliable system with increased redundancy. It has four mechanical trains for safety injection, direct-to-vessel injection connections, and an in-containment refueling water storage tank. The same size pumps and valves used in the original System 80 two train design are now used in all four trains. The trains are not interconnected by common headers and include provision for full flow, on-line testing to eliminate the need to extrapolate bypass-flow test results to demonstrate compliance to Technical Specifications (Section 6.3.2).

- In-Containment Refueling Water-Storage Tank (IRWST): The IRWST has been located in the containment building, in a torus-like configuration around the reactor vessel cavity. Containment water collection points empty into the IRWST. This means that the safety injection pumps always take water from the tank, eliminating the need to switch from tank to containment sump following a loss of coolant accident (Section 6.8).

- Safety Depressurization System (SDS): The SDS is a dedicated manually-operated system designed to permit depressurization of the Reactor Coolant System (RCS) when normal processes are not available. The SDS provides the capability to rapidly depressurize the RCS so that an operator can initiate primary system feed and bleed (using the safety injection pumps) to remove decay heat following a total loss of feedwater event. Manual control of motor operated valves enable discharge from the pressurizer to be directed to the IRWST, without the unreliability concern that is associated with automatically operating valves (Section 6.7).

- Emergency Feedwater System (EFWS): The EFWS is a dedicated safety system intended for emergency use only. (The Main Feedwater System includes a startup pump and a full range control system for normal startup and shutdown operations).

  The EFWS has two separate trains. Each consists of one emergency feedwater storage tank, one full capacity motor-driven pump, one full capacity non-condensing turbine-driven pump, and one cavitating venturi. The cavitating venturi minimizes excessive emergency feedwater flow to a steam generator with a ruptured feed or steam line. The EFWS therefore requires no provision for automatic isolation of emergency feedwater flow to a steam generator having a ruptured steam line or feed line (Section 10.4.9).

- Shutdown Cooling System (SCS): The SCS design pressure has been increased to 900 psig. This higher pressure provides greater operational flexibility and eliminates concern for system over-pressurization. The SCS is interconnected with the Containment Spray System, which uses identical pumps. The reliability of both systems is therefore increased, and each set of pumps can serve as a backup for the other (Section 5.4.7).

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5. Plant Structures and Arrangements:

- The containment for System 80+ is a 200-foot diameter steel sphere which maximizes space for equipment and maintenance while minimizing unusable volume in the upper part of the containment. The operating floor offers 75% more usable area than a cylindrical containment of equal volume (Sections 3.8 and 6.2).

- Features for mitigating the consequences of postulated severe accidents include a reactor vessel cavity designed to improve the ability to resolidify molten core material on the cavity floor by cooling and retaining the molten core debris (Section 6.8).

- The spherical containment provides a lower annulus under the sphere which replaces a conventional safety-grade auxiliary building, and is an ideal location for safety systems. Placing of the safeguards equipment in the subsphere areas is an economically attractive approach to addressing numerous regulations associated with this equipment. Separation for internal flood mitigation, fire protection, security, and sabotage concerns are easily addressed without adverse affect on accessibility (Section 3.8).

Question 210.3

It is not always apparent on the P&ID's presented in the CESSAR-DC what the safety class is of a particular portion of a system. If no class changes are shown on a P&ID, is this because it is to be assumed that the entire system is designated as being the same safety class as given in Table 3.2-1? If so, then a note should be added to each sheet of the P&ID stating the safety class of that system. Also, where one system interfaces with another system, the safety class changes should be clearly identified on the P&ID for each interfacing system. For example, where is the safety class change shown for the Safety Injection System interface to the Reactor Coolant System in the Direct Vessel Injection mode where the SIS piping ties directly to the reactor vessel? There should be a safety class change from SC-2 to SC-1 shown on the P&ID for both the Reactor Coolant System and the Safety Injection System (Ref. Figure 5.1.2-1 and Figure 6.3.2-1B). Revise all P&ID's for safety-related systems to include this information.

Response 210.3

All P&ID's are being reviewed to ensure safety class is specified. Safety class and safety class changes will be noted on the P&ID's or a note will be added to clarify [e.g., all piping and components on this drawing are non-nuclear safety (NNS), or all piping and components on this drawing are Safety Class 3(SC-3) unless shown otherwise, etc.].

In addition, the P&ID legend drawing for CESSAR-DC (Figure 1.7-1) will be revised to clarify that a safety class designation of 4 corresponds to non-nuclear safety (NNS).

Some P&ID's in CESSAR-DC already show the safety classes, and this information was added to some P&ID's in answering other RAIs. The attached table lists all CESSAR-DC P&ID's and flow diagrams and the status of whether or not the safety classes and safety class changes are shown. Note that some of these flow diagrams are not currently in CESSAR-DC, however they will be added in a future amendment. The P&ID's and flow diagrams that do not currently show safety classes will be updated in a future CESSAR-DC amendment to specify the safety classes.

For the example cited above (DVI to reactor vessel), the safety classes and safety class changes are shown in CESSAR-DC, Amendment I, Figures 5.1.2-1 and 6.3.2-1C. Safety Class 1 (SC-1) is shown for all DVI piping in Figure 5.1.2-1, and the safety class change from SC-1 to SC-2 is shown in Figure 6.3.2-1C.

Response 210.3 (Cont'd)

CESSAR-DC FIGURE	SYSTEM	SAFETY CLASS DESIGNATION STATUS
5.1.2-1	Reactor Coolant System	Yes, CESSAR-DC
5.1.2-2	Reactor Coolant Pump	Yes, CESSAR-DC
5.1.2-3	Pressurizer and Safety Depressurization System	Yes, CESSAR-DC
6.2.3-1	Annulus Ventilation System	No
6.2.5-1	Containment Hydrogen Recorbiner System	No
6.3.2-1A	Safety Injection System (includes Shutdown Cooling and Containment Spray Systems)	Yes, CESSAR-DC
6.3.2-1B	Safety Injection System (includes Shutdown Cooling and Containment Spray Systems)	Yes, CESSAR-DC
6.3.2-1C	Safety Injection System (includes Shutdown Cooling and Containment Spray Systems)	Yes, CESSAR-DC
6.8-4	In-Containment Water Storage System	Yes, CESSAR-DC
9.1-3	Pool Cooling and Purification System	Yes, CESSAR-DC
9.2.1-1	Station Service Water System	Yes, RAI 410.110
9.2.2-1	Component Cooling Water System	Yes, CESSAR-DC
9.2.3-1	Demineralized Water Makeup System	No
9.2.6-1	Condensate Storage System	No
9.2.9-X	Essential and Normal Chilled Water Systems	Yes, RAI 410.113
9.3.1-1	Instrument Air System	No
9.3.1-2	Station Air System	No
9.3.1-3	Breathing Air System	No
9.3.2-X	Primary Sampling System	No

Response 210.3 (Cont'd)

CESSAR-DC FIGURE	SYSTEM	SAFETY CLASS DESIGNATION STATUS
9.3.2-X	Secondary Chemistry Control System	No
9.3.3-1	Containment Building Floor Drain System	No
9.3.3-2	Reactor Building Subsphere Floor Drain System	No
9.3.3-3	Nuclear Annex Radioactive Floor Drain System	No
9.3.3-4	Nuclear Annex Nonradioactive Floor Drain System	No
9.3.3-5	Nuclear Annex Radioactive Equipment Floor Drain System for CVCS Equipment	No
9.3.4-1	Chemical and Volume Control System	Yes, CESSAR-DC
9.4-2	Nuclear Annex Control Building Air Flow Diagram	No
9.4-3	Fuel Building Air Flow Diagram	No
9.4-4	Reactor Building Subsphere Cooling Air Flow Diagram	No
9.4-5	Reactor Building Subsphere Ventilation Air Flow Diagram	No
9.4-6	Containment Cooling Purge and Pressure Control Air Flow Diagram	No
9.4-X	Diesel Building Ventilation System	No
9.4-X	Nuclear Annex Ventilation System	No
9.5.1-1	Fire Protection Water Distribution System	No
9.5.4-1	Diesel Generator Engine Fuel Oil System	No
9.5.5-1	Diesel Generator Engine Cooling Water System	No

Response 210.3 (Cont'd)

CESSAR-DC FIGURE	SYSTEM	SAFETY CLASS DESIGNATION STATUS
9.5.6-1	Diesel Generator Engine Starting Air System	No
9.5.7-1	Diesel Generator Engine Lube Oil System	No
9.5.8-1	Diesel Generator Engine Intake and Exhaust System	No
9.5.9-1	Diesel Generator Building Sump Pump System	No
10.1-2	Main Steam and Feedwater System	Yes, CESSAR-DC
10.3.2-1	Main Steam and Extraction Steam Systems	No
10.4.2-1	Main Condenser Evacuation System	No
10.4.2-2	Main Condenser Evacuation System	No
10.4.5-1	Condenser Circulating Water System	No
10.4.7-1	Condensate, Feedwater and Heater Drain Systems	No
10.4.8-1	Steam Generator Blowdown System	No
10.4.9-1	Emergency Feedwater System	Yes, CESSAR-DC
11.2-1	Liquid Waste Management System	No
11.3-1	Waste Gas Management System	No
11.4-1	Solid Waste Management System	No

Question 210.87:

GSI B-55, concerning improved reliability of Target Rock safety relief valves, is categorized as a "drop" issue in Table A1-1 of CESSAR-DC Appendix A under a generic explanation of Category 1.f. Since staff evaluation on that issue is still proceeding and a resolution of that issue has not yet been concluded, provide either your basis to justify the closure of this issue, or verify that the issue is not applicable to the CE Standard System 80+.

Response 210.87:

GSI B-55 addresses the improved reliability of Target Rock Power Operated Safety/Relief valves in BWR's. The category assigned to this GSI is 1b and not 1-f as stated above. Category 1b events are events which are specific to another plant design, (e.g., BWR, W, B&W). It is our opinion that this Generic Safety Issue is not applicable to C-E plants as we do not use Target Rock Power Operated Safety/Relief valves on the reactor coolant system, the charging system or the safety injection systems. Therefore, we believe that our categorization of this event is correct and that this issue is not applicable to System 80+ design.

Question 220.6

Section 3.7.1.1 - Explain why Figure 3.7-2 is practically identical to Figure 3.7-1. In Case A-1 (Figures 3.7-1 and 3.7-2), the bedrock is at the foundation level. Explain why, in the high frequency range, the horizontal and vertical spectral values converge to 0.5g and 0.3g, respectively in both the cases.

Response 220.6

Figure 3.7-2 was erroneously presented. The correct figure is being prepared and will be included in a future amendment to CESSAR-DC.

Question 220.7

Section 3.7.1.1 - This section states that "For the time history method of analysis, three design time histories are generated that are consistent with the design rock outcrop spectra at the free field." Explain why these time histories instead of the time histories consistent with the spectra presented in Section 3.7.1.1 are used in the time history method of analysis. Also, compare the PSDs of the 3.7.1.1 spectra with the provisions of Appendix A of SRP 3.7.1.

Response 220.7

The rock outcrop time histories were used only in the fixed-base analysis. The SSI analyses utilized as control motions the response time histories of the soil at the free-field ground surface. These are the time histories that produce the unsmoothed response spectra shown in Figures 3.7-1 to 3.7-24.

CESSAR-DC, Section 3.7.1.2 will be revised in a future amendment to read as follows:

3.7.1.2 Design Time History

"Since the System 80+ Standard Design is designed for generic site conditions, for the time history method of analysis, the generic free-field ground surface time histories are used as control motions in the analyses. In the soil-structure interaction analyses, for each generic site, the corresponding two horizontal and one vertical time histories at the free-field ground surface are used with the SSI model of that site. These time histories produce the unsmoothed response spectra shown in Figures 3.7-1 to 3.7-24. For the fixed-base analysis, the rock outcrop time histories are directly used as the control time histories. The response spectra at 5% damping corresponding to the rock outcrop time histories are shown in Figures 3.7-25 to 3.7-27."

RAI 220.7

3.7 SEISMIC DESIGN

3.7.1 SEISMIC INPUT

3.7.1.1 Seismic Input

This section discusses the seismic design parameters and methodologies being used for the design of those systems and subsystems important to safety and classified as Category I in Section 3.2.

The System 80+ Standard Design as defined by CESSAR-DC is not based on a specific site. Generic site conditions were selected to cover a range of possible conditions for the System 80+ sites. More specifically, sets of representative cases from each of four generic site categories were evaluated to create the ground surface and foundation level spectra shown in Figures 3.7-1 through 3.7-24. Out of 12 soil cases analyzed in Section 2.5.2, nine are used in the soil structure interaction (SSI) analysis. The three cases eliminated in the SSI analysis (A1, B3 and D1) were non-governing cases whose soil response levels were enveloped by other cases. See Section 2.5.2 for details of this analysis phase.

The effect of differential seismic displacement on the equipment and supports is included in the analysis as described in Section 3.7.2.1. I

3.7.1.2 Design Time History

Insert →

Delete

~~For the time history method of analysis, three design time histories are generated that are consistent with the design rock outcrop spectra at the free field. The characteristics of each time history are presented in Section 2.5.2.5.1. The response spectra plots for these time histories are shown in Figures 3.7-25 through 3.7-27.~~

3.7.1.3 Critical Damping Values

Damping values used for various nuclear safety-related structures systems and components are based upon Regulatory Guide 1.61 or ASME Code Case N-411-1 (See Figure 3.7-41). These values are expressed in percent of critical damping and are given in Table 3.7-1. When the response spectra method of analysis is used for piping, damping values are based on Code Case N-411-1.

RAI 220.7

Insert

Since the System 80+ Standard Design is designed for generic site conditions, for the time history method of analysis, the generic free-field ground surface time histories are used as control motions in the analyses. In the soil-structure interaction analyses, for each generic site, the corresponding two horizontal and one vertical time histories at the free-field ground surface are used with the SSI model of that site. These time histories produce the unsmoothed response spectra shown in Figures 3.7-1 to 3.7-24. For the fixed-base analysis, the rock outcrop time histories are directly used as the control time histories. The response spectra at 5% damping corresponding to the rock outcrop time histories are shown in Figures 3.7-25 to 3.7-27.

QUESTION 220.55

SECY-90-016 Issues in CESSAR-DC; Issues 1 and 11 - Public Safety Goals and Containment Performance.

Section 5.4 Appendix B addresses the staff guidance of Containment Conditional Failure Probability (CCFP) of 0.1. The SRM related to SECY-90-016 also recommends the use of deterministic objectives. As such, the staff (ESGB) would prefer a deterministically established containment performance objective, such as establishing containment structural failure criteria based on the maximum strains (in general shell and in localized areas) as limited by global restraints and functionality of various featuresrelated to the mitigation of the consequences of accidents. Provide information regarding the ultimate capacity ofvthe containment considering the above attributes. (see also RAI 720.24 in Section 7 of Appendix B).

RESPONSE 220.55

Combustion Engineering agrees that the staff position on Issue 11 of SECY-90-016 permits a demonstration of adequate containment performance by either a 0.1 Containment Conditional Failure Probability (CCFP) criterion or by a deterministic goal that offers equivalent protection. It is our understanding that the staff is concerned with the use of the 0.1 CCFP criterion because of the uncertainties inherent in the PRA methods, especially those associated with seismic hazards. Therefore, C-E is investigating the use of an alternate, deterministic criterion to gain additional insight into System 80+ containment performance for severe accident conditions.

For this investigation, the choice of the severe accident sequences and the assumptions on degraded core behavior will be based on best-estimate judgments which will be consistent with the MAAP analysis reported in the PRA documented in Appendix B of CESSAR-DC. The purpose of this analysis is to determine the time response of the containment pressure and temperature for a representative severe accident scenario. Using this information, the length of time the containment pressure remains below the ASME Service Level C criterion is determined. For the System 80+ containment, the ASME service Level C criterion is 141.5 psia at a containment temperature of 350 degrees F.

Using the above guidelines, a station blackout scenario was simulated using the MAAP computer code. With battery power available, auxiliary feedwater flow is assumed to be provided by the turbine driven auxiliary feedwater pump for a period of eight hours. Following the unavailability of auxiliary feedwater at eight hours, the vessel fails at 15.5 hours due to the loss of core cooling. Cavity flooding is assumed to occur prior to vessel failure. Combined with the deentraining characteristic of the System 80+ cavity, this will result in the retention of approximately 87.5% of the corium within the cavity. For this scenario, the containment pressure remains below 100 psia for up to 50 hours after the initiation of the station blackout (Please see attached Figure 1). The containment temperature does not exceed 350 degrees F during this time frame (Please see Figure 2). These results demonstrate that, for this best estimate severe accident scenario, the containment pressure remains well below ASME Service Level C criterion of 141.5 psia for more than 50 hours.

The ultimate pressure capacity for the System 80+ containment is approximately 185 psia at a temperature of 290 degree F. This value is not considered specifically in the above evaluation, but is included here to demonstrate the additional margin to System 80+ containment integrity during a severe accident.

FIGURE 1

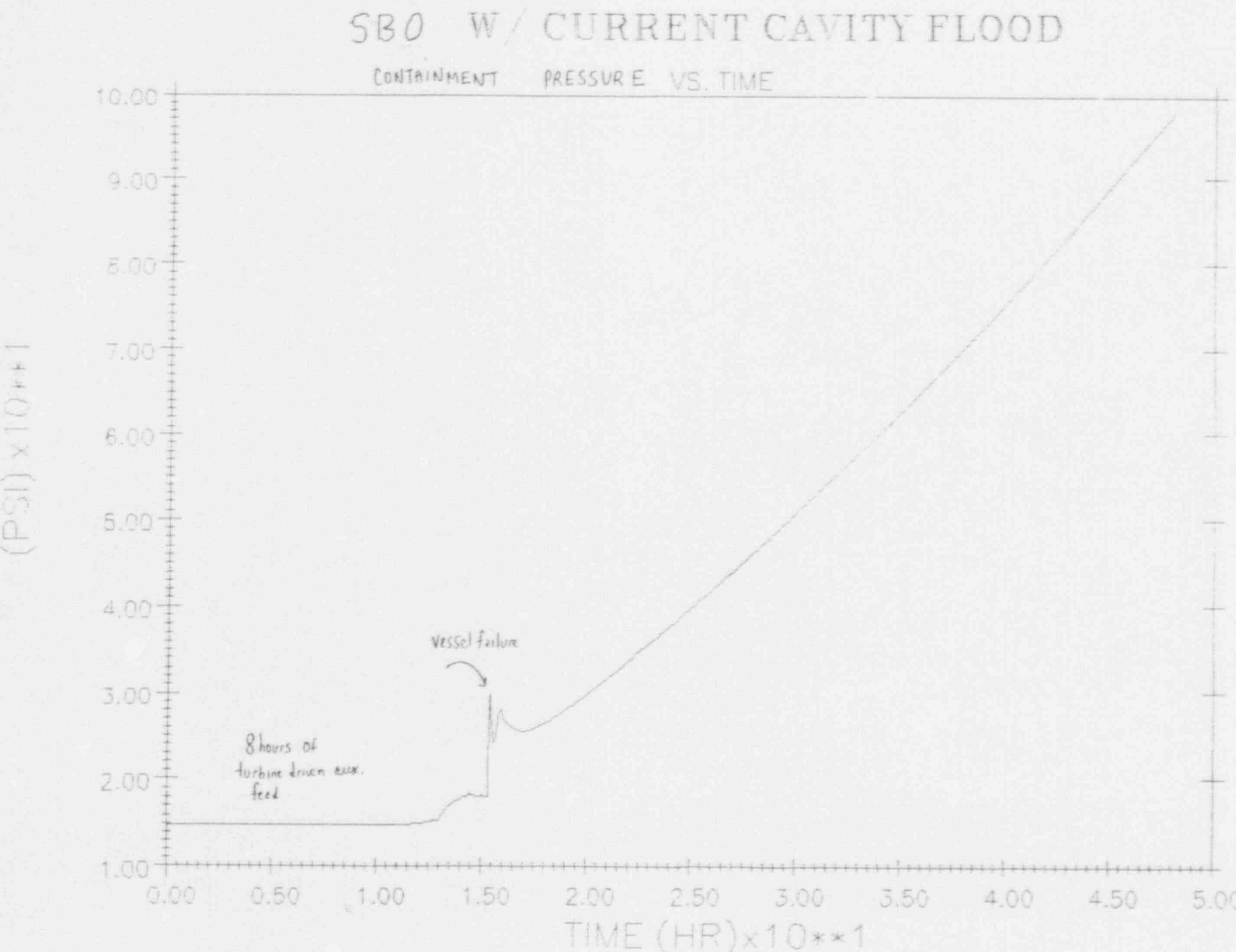
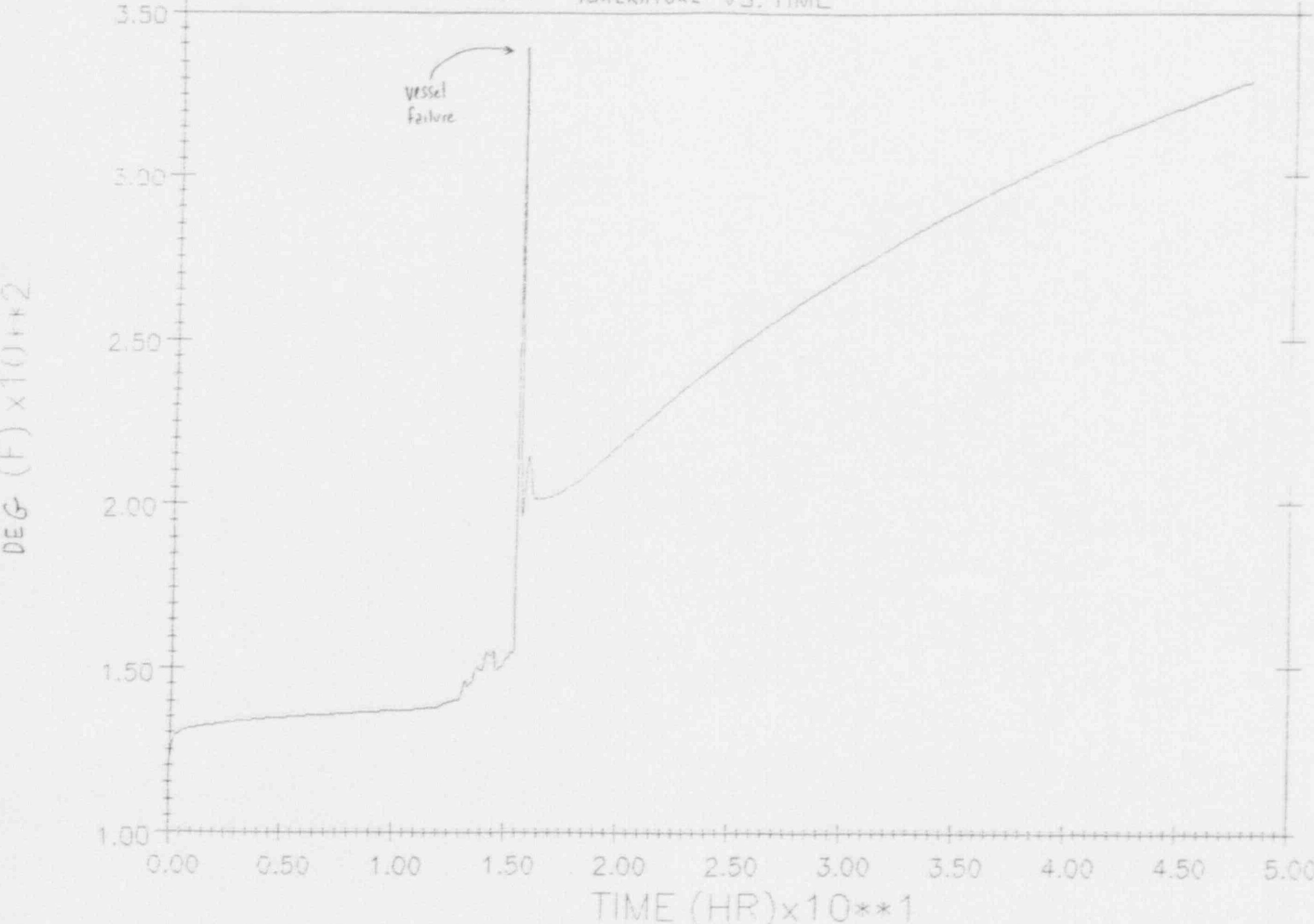

SBO W/ CURRENT CAVITY FLOOD
CONTAINMENT PRESSURE VS. TIME
(PSI) x10**1
10.00
9.00
8.00
7.00
6.00
5.00
4.00
3.00
2.00
1.00
0.00
0.50
1.00
1.50
2.00
2.50
3.00
3.50
4.00
4.50
5.00
TIME (HR) x10**1
8 hours of
turbine driven aux.
feed
vessel failure

FIGURE 2

SBO W/ CURRENT CAVITY FLOOD

CONTAINMENT TEMPERATURE VS. TIME



Question 230.4

Section 2.5.2.5.1 - The text states that the spectrum is based on work discussed in Reference 1. Provide Reference 1 (P. 2.5-10) and expand the text to explain the basis for spectrum. Include in your explanation the following points:

- The EPRI-ALWR RD presumes a RG 1.60 spectrum with a ZPA of 0.3g.

- The rock outcrop spectrum for eastern North America shown in Figure 2.5-5 is rich in high frequency content. Explain why the frequency content above 50 Hz is not included in the selected horizontal spectrum.

  The text states that the spectral ordinates are based on using NUREG-0098; however, the spectral ordinates shown in Figure 2.5-5 are less than those obtained using NUREG-0098 for frequencies lower than 4 Hz.

- Although an SSE with a peak horizontal acceleration of 0.30g at the ground surface is adequate for most sites in the United States, it may not envelope the ground motion for sites near seismically active areas in the eastern and central United States or sites in the western United States.

- The text states the vertical component is equal to two-thirds of the horizontal spectrum at all frequencies. However, Revision 2 (dated August 1989) of Standard Review Plan Section 2.5.2 text states that both horizontal and vertical site-specific response spectra should be developed. The staff has questioned the use of the ratio two-thirds because for most plants the controlling earthquake for the SSE is one that is assumed to be near the site and there is data from recorded nearby earthquakes which show that the vertical and horizontal ground motions can be of similar amplitude. At the October 1990 meeting with the staff, CE's consultant had stated that for rock sites the vertical ground motion spectra would be made equal to the horizontal ground motion spectra, while for the soil sites the vertical spectra would remain equal to 2/3 of the horizontal spectra. It is not clear how this suggestion has been implemented in the DC documents.

Response 230.4

Reference 1 is provided in Draft form (See response to RAI 220.4). In addition, the following information is provided.

°RG 1.60 Spectrum

The System 80+ target spectrum at a rock outcrop was based on enveloping two spectra, each anchored to a ZPA of 0.3g. One spectrum was derived using the random vibration theory (RVT) for conditions representative of the seismic characteristics for eastern North America earthquake ground motions. The other spectrum was derived using the spectral shape proposed by Newmark and Hall in NUREG-0098. A spectral amplification of 1.0 was used for frequencies above 40 Hz because these frequencies are outside the range of engineering interest.

It should be noted that the System 80+ target spectrum was used as input rock motion to the soil profiles considered in CESSAR-DC. The motion calculated at the ground surface was then used for the evaluation of soil-structure interaction and for the evaluation of the structures, components and equipment. The System 80+ target spectrum was used directly in the structural evaluations only for the fixed-base cases.

The spectral ordinates calculated at the ground surface of all the soil profiles considered are shown in Figure 1 together with the RG 1.60 spectrum anchored to 0.3 g. As can be noted, the spectra at the ground surface of the soil profiles are equal to or significantly exceed the RG 1.60 spectrum for frequencies higher than about 0.7 Hz. Similarly, as shown in Figure 2, the spectral ordinates at the foundation level are equal to or significantly exceed 0.6 times RG 1.60 spectrum.

Thus, the selected System 80+ target spectrum, together with the methodology used in evaluating structures, compoennts and equipment provide a more conservative approach than using the RG 1.60 spectrun directly as input in the evaluation.

Cut-off of Frequencies above 40 Hz

There are two main reasons that frequency content above 40 Hz is not included in the rock outcrop response spectra:

1. Industry studies have demonstrated that seismic motions rich in high frequency content are not damaging to structures and equipment with even minimum ductility (Reference 230.4.1). The rock outcrop spectra used in the System 80+ seismic analysis are rich in frequencies up to 25 Hz, which is well above the cutoff values recommended in Reference 230.4.1 for typical Eastern North America spectra.

Response 230.4 (continued)

2. In many cases, including frequency content above 25 Hz causes accurate SSI and other civil/structural analyses to be nearly impossible to obtain. The level of detail provided in analytical models according to current practice is not sufficient for analyses with frequencies above 25 Hz. Analytical models constructed according to state-of-the-practice procedures cannot capture effects of high-frequency motions.

° Use of NUREG-0098

As noted in CESSAR-DC, it is intended that the ZPA be based on an 84th percentile acceleration not exceeding 0.3 g. Therefore, to obtain 84th percentile spectral ordinates it is sufficient to multiply the 84th percentile ZPA by the median spectral shape. This is illustrated in Figure 3 which shows the spectral shapes based on either dividing the median spectral values by the median ZPA or by dividing the 84th percentile spectral values by the 84th percentile ZPA. The two procedures give almost identical results except at a few frequencies as shown in Figure 4, with the maximum difference being about 15 percent. Note that the spectral values used for Figures 3 and 4 were obtained from spectral analysis of 28 horizontal accelerograms at rock sites during the 1989 Loma Prieta earthquake. These records were recorded at distances ranging from about 5 km to 80 km, and had peak accelerations ranging from about 0.04 g to 0.64 g.

Accordingly, the System 80+ target spectrum was derived in part by using the median spectral shape from NUREG-0098 anchored to 0.3 g.

° SSE Peak Acceleration

If the SSE peak acceleration at a given location is greater than 0.3 g, then site-specific evaluations are required.

° Vertical Component

The System 80+ target vertical spectrum at a rock outcrop was chosen equal to 2/3 the horizontal spectrum and used in the analyses as input vertical rock motion to the soil profiles. The resulting spectrum of the calculated vertical motion at the ground surface for a few of the cases was lower than the horizontal spectrum. For most of the cases, however, the spectrum of the calculated vertical motion at the ground surface equaled or exceeded the horizontal spectrum over a significant range of frequencies.

Response 230.4 (continued)

These variations are illustrated in Figures 5 through 7. Figure 5 shows the spectra for the horizontal components and for the vertical component calculated at the ground surface for Case B-1. The vertical spectral ordinates are smaller than the horizontal spectral ordinates for all frequencies. Figures 6 and 7 show corresponding results for Cases C-3 and C-1.5, respcetively. The vertical spectral ordinates for Case C-3 are signifcantly larger than the horizontal spectral ordinates for frequencies higher than about 6 Hz and in the frequency range of about 1.6 to 3 Hz. For Case C-1.5, the vertical spectral ordinates are comparable to the horizontal spectral ordinates for frequencies higher than about 2.5 Hz.

The ratio of the vertical spectral ordinate divided by the average spectral ordinates of the two horizontal compoents is shown in Figure 8 for all the cases considered. Also shown in Figure 8 are the median values of this ratio for the entire frequency range. As can be noted in Figure 8, the median values of this ratio are equal to or greater than one for most of the cases considered at frequencies higher than about 4 Hz. In fact, the values of this ratio are greater than one over the frequency range of 1 to 100 Hz for all cases considered except for Cases A-1, B-1 and C-1 and for Case B-1.5 over the frequency range of 1 to 3 and 5 to 100 Hz as listed below:

	Ratio of Vertical Spectral Ordinates Divided by the Average of the Two Horizontal Spectral Ordinates over the Frequency Range of			
Case No.	1 to 3 Hz	3 to 5 Hz	5 to 10 Hz	10 to 100 Hz
Case A-1	< 1	< 1	< 1	< 1
Case B-1	< 1	< 1	< 1	< 1
Case B-2	> 1	> 1	> 1	> 1
Case B-3	> 1	> 1	> 1	> 1
Case B-4	> 1	> 1	> 1	> 1
Case C-1	< 1	< 1	< 1	< 1
Case C-2	> 1	> 1	> 1	> 1
Case C-3	> 1	> 1	> 1	> 1
Case D-1	> 1	> 1	> 1	> 1
Case B-1.5	< 1	> 1	< 1	< 1
Case B-3.5	> 1	> 1	> 1	< 1
Case C-1.5	> 1	> 1	> 1	> 1

Response 230.4 (continued)

Thus, the vertical component has been adequately incorporated in the development of earthquake ground motions at the ground surface of all the soil cases to reflect the possibility that the vertical and the horizontal spectral ordinates can be of comparable amplitude over the higher frequency ranges.

The spectral ordinates at the foundation level for all of the soil profiles considered also generally envelope the System 80+ horizontal target spectrum. Therefore, we believe that System 80+ is adequately designed for rock sites with the vertical design spectrum defined as being equal to the horizontal design spectrum.

Reference 230.4.1: Reed, J.W., Kennedy, R.P., Lashkari, B., Kassawara, R.P., "Analysis of High-Frequency Seismic Effects", Proceedings, Third Symposium on Current Issues Related to Nuclear Power Plant Structures, Equipment and Piping, Orlando, Florida, December 1990.

RAI 230.4

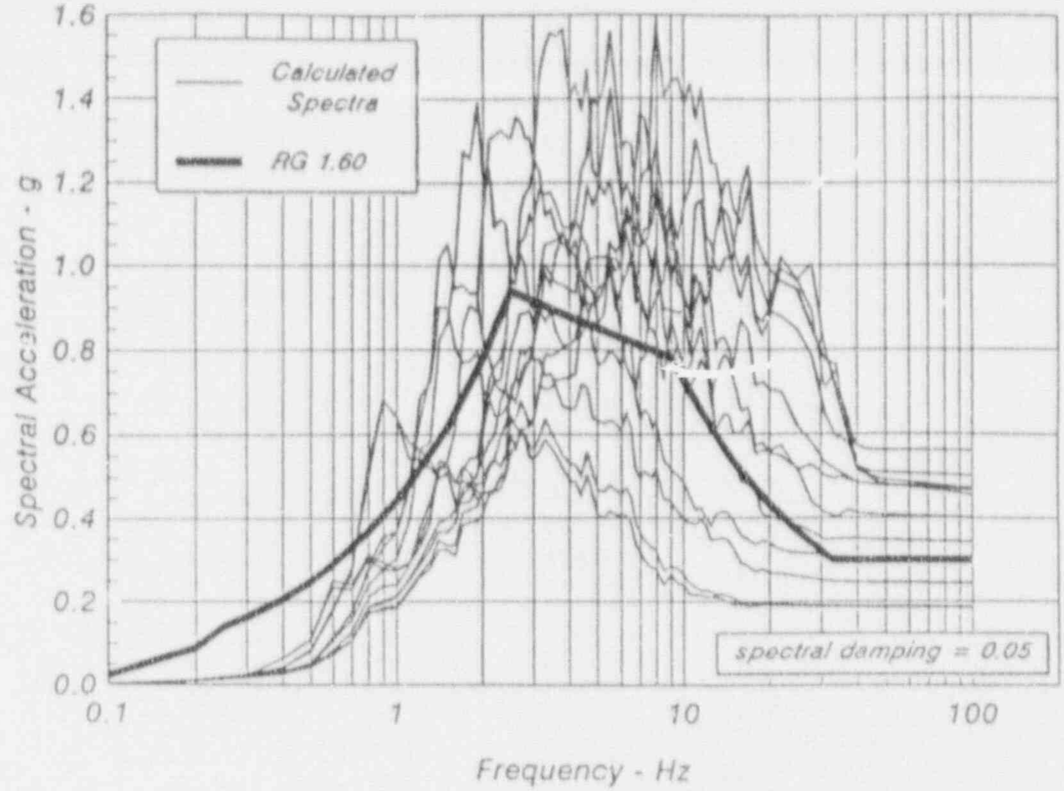


Fig. 1 Comparison of Spectral Ordinates Calculated at the Ground Surface with RG 1.60 Anchored to 0.3 g

IMI - 12/30/91

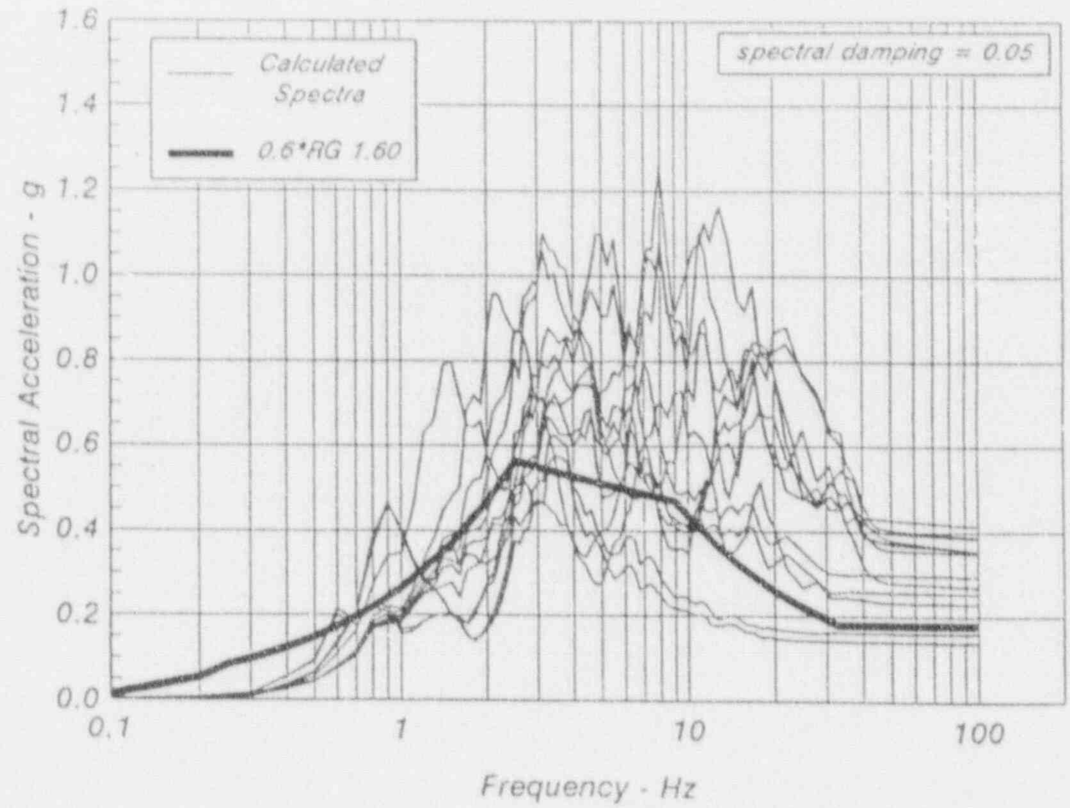


*Fig. 2 Comparison of Spectral Ordinates Calculated at the Foundation Level with RG 1.60 and with 0.6*RG 1.60 Anchored to 0.3 g*

IMI - 12/30/91

RAI 230.4

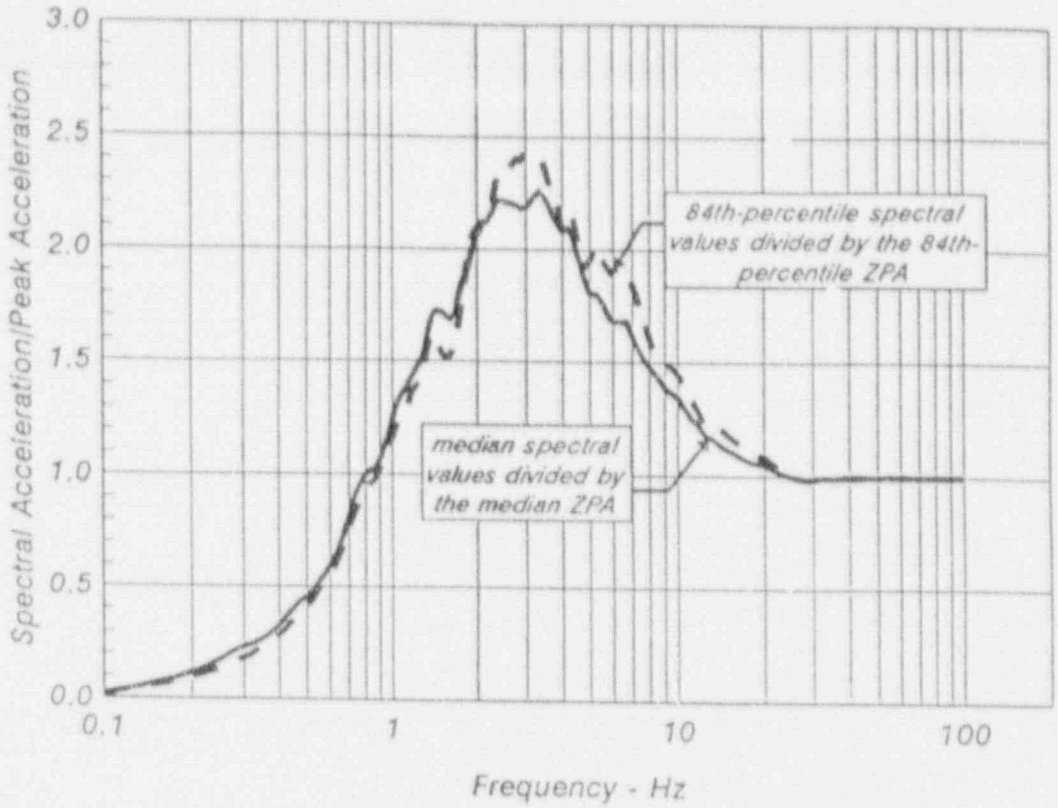


Fig. 3 Spectral Shapes Based on Dividing the Median Spectral Values by the Median ZPA or by Dividing the 84th Percentile Spectral Values by the 84th Percentile ZPA

IMI - 12\30\91

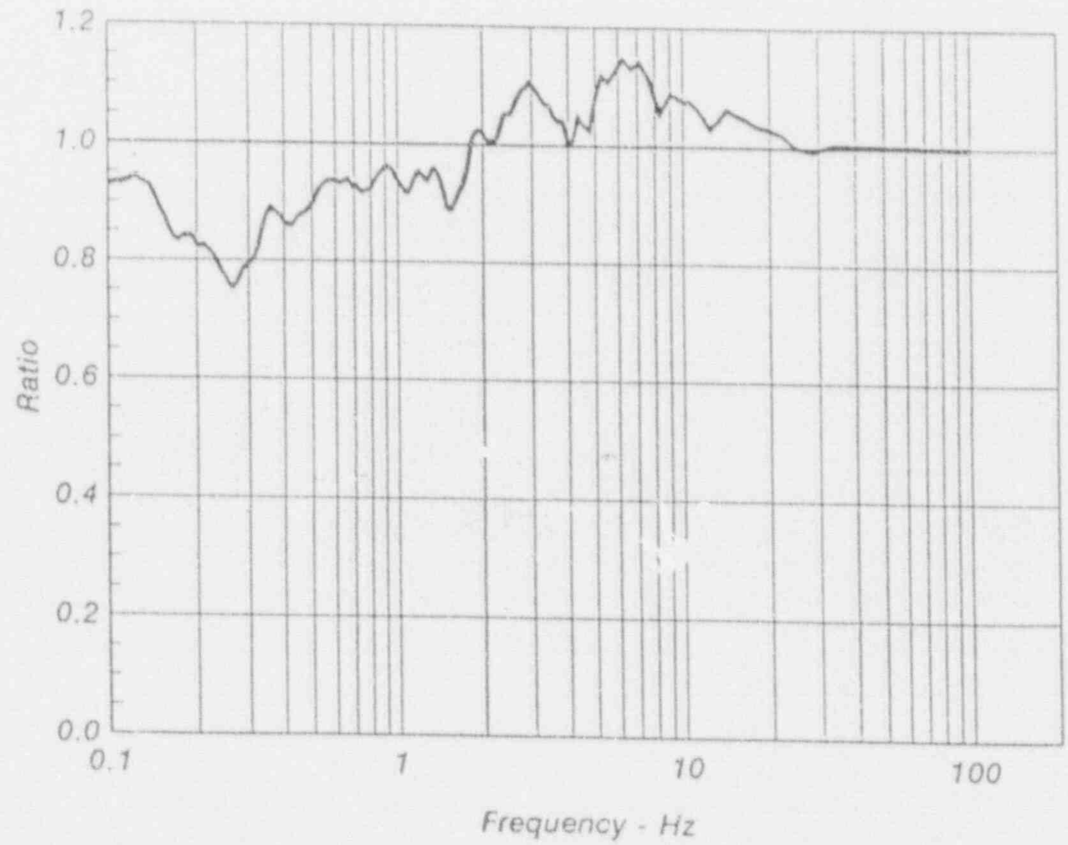


Fig. 4 Ratio of the Two Spectral Shapes Shown in Figure 3

IMI - 12\30\91

RAI 230.4

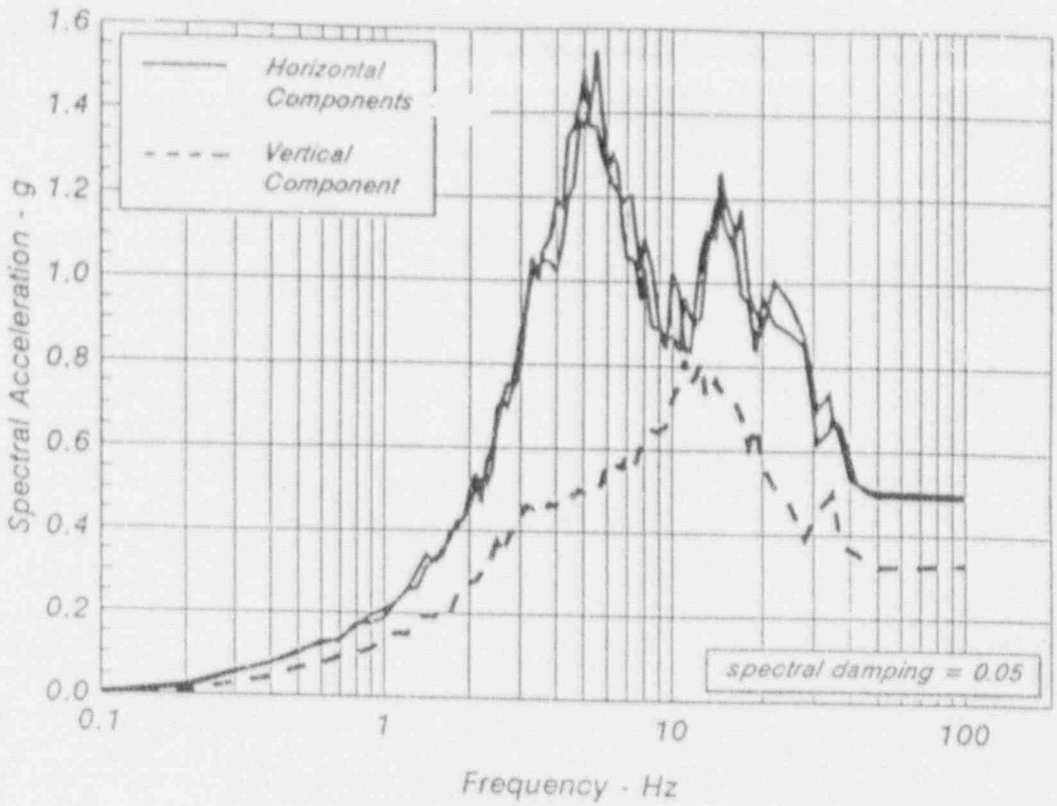


Fig. 5 Calculated Horizontal and Vertical Spectra at the Ground Surface for Case B-1

IMI - 12|30|91

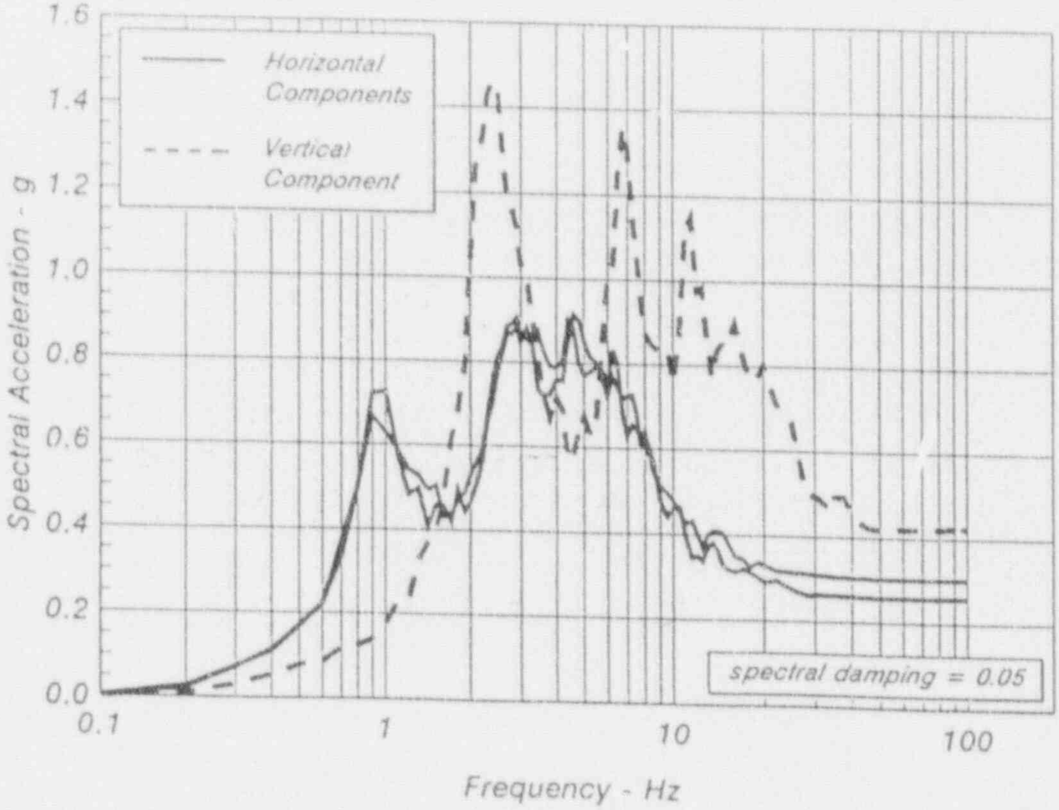


Fig. 6 Calculated Horizontal and Vertical Spectra at the Ground Surface for Case C-3

IMI - 12|30|91

RAI 230.4

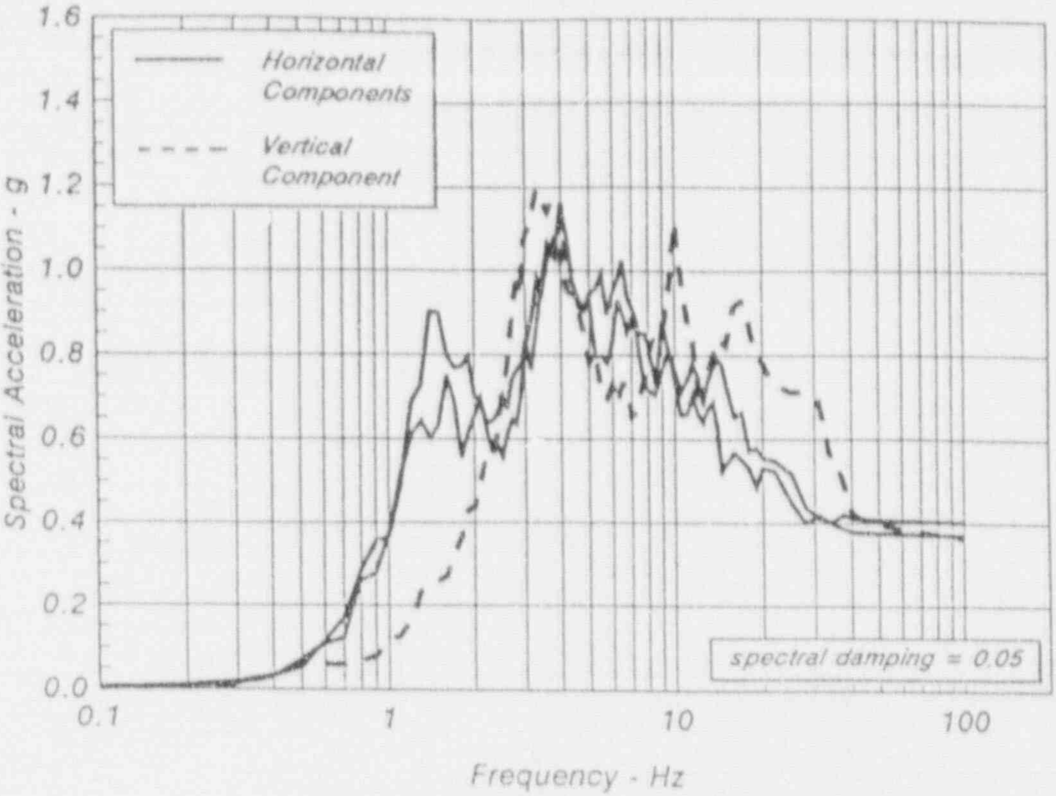


Fig. 7 Calculated Horizontal and Vertical Spectra at the Ground Surface for Case C-1.5

IMI - 12/30/91

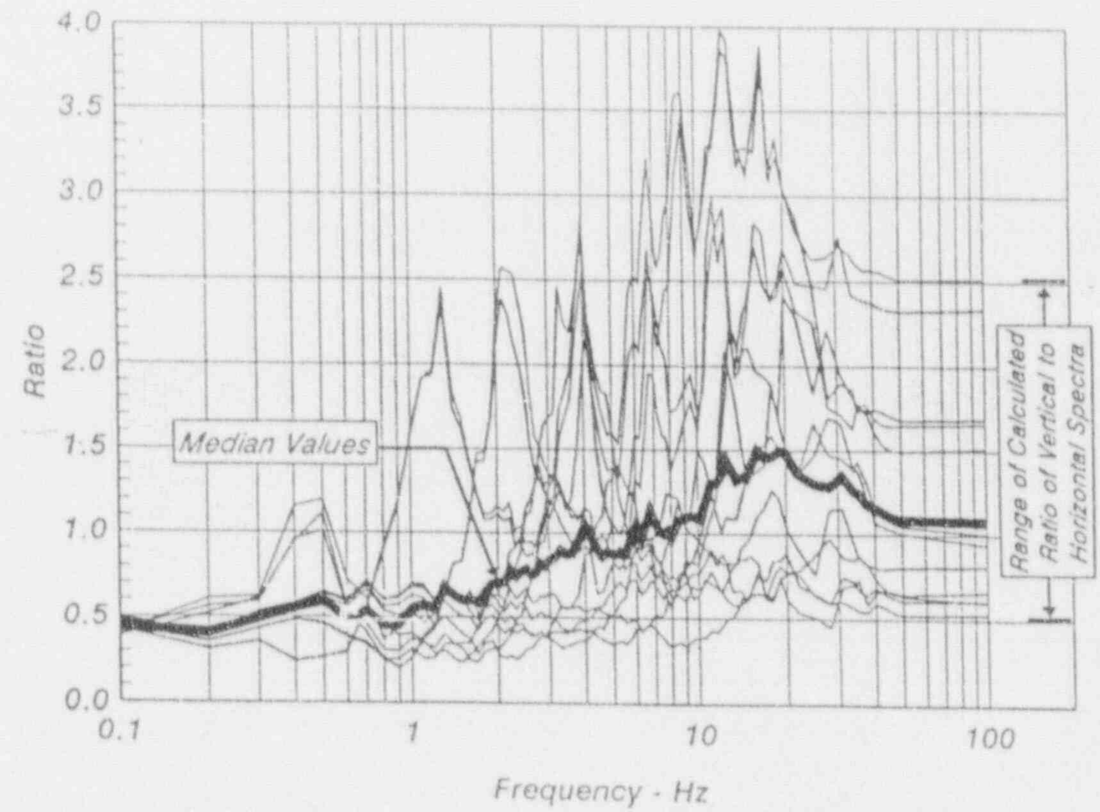


Fig. 8 Ratio of the Calculated Spectral Ordinates for the Vertical Component Divided by the Calculated Spectral Ordinates for the Horizontal Components

IMI - 12/30/91

Question 230.10

Appendix 2A

- The LLNL seismic hazard analysis uses spectra. accelerations for different soil categories than those shown on Figure 2A-I. Is the use of different soil categories significant?

- There should be a case of soil with a Vs of about 1000 ft./sec. over a rock with a Vs of at least 6000 ft./sec. (see November 13, 1990, Summary of CE System 80+ meeting). Discuss the basis for not considering such a case.

Response 230.10

The results of the cases analyzed as part of the System 80+ design and described in CESSAR-DC would cover the results that would be obtained using the range of properties used in the LLNL seismic hazard analysis. Therefore we believe that the range of properties and site categories used for CESSAR-DC are adequate and would cover those selected by LLNL.

In response to this question, a soil layer 52 ft. deep (designated A-0) and a soil layer 100 ft. deep (designated B-0) each having a maximum shear wave velocity v_s = 1,000 fps and overlying a rock half-space whose velocity is equal to 6,000 fps were selected for evaluation. The seismic response of each layer to the synthetic time history H1 was calculated and the results are presented in Figs. 1 and 2. Figure 1 shows the spectral ordinates calculated at the ground surface and at the foundation level for Case A-0; similar plots for Case B-0 are presented in Fig. 2. The results for these cases are compared in Figures 3 and 4 to the spectral ordinates calculated for the previous cases. Figure 3 shows the results at the ground surface and Fig. 4 shows the results at the foundation level. Both figures indicate that the results for these two cases are within the range of the previously calculated cases except for Case A-0 at the ground surface in the frequency range of about 2.6 to about 3.8 Hz. Over this frequency range, the spectral ordinates at the ground surface for Case A-0 exceed the range of the previously calculated cases by no more than about 25 percent.

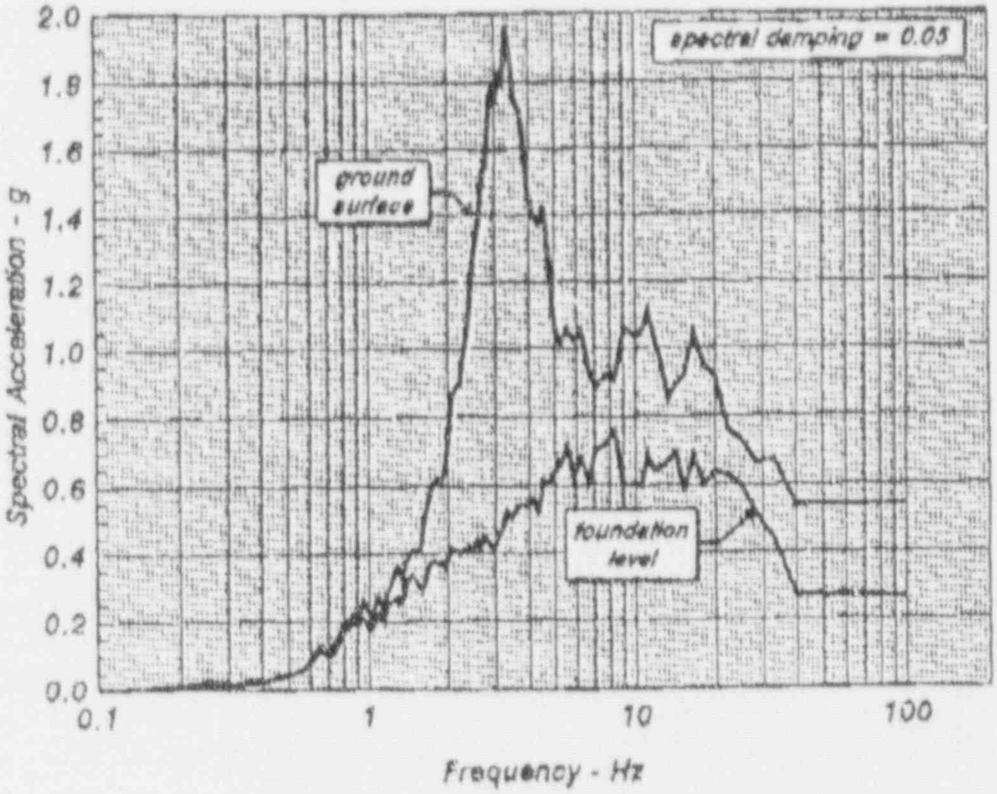


Fig. 1 Spectral Ordinates Calculated for Case A-0

IMI - 07|13|82

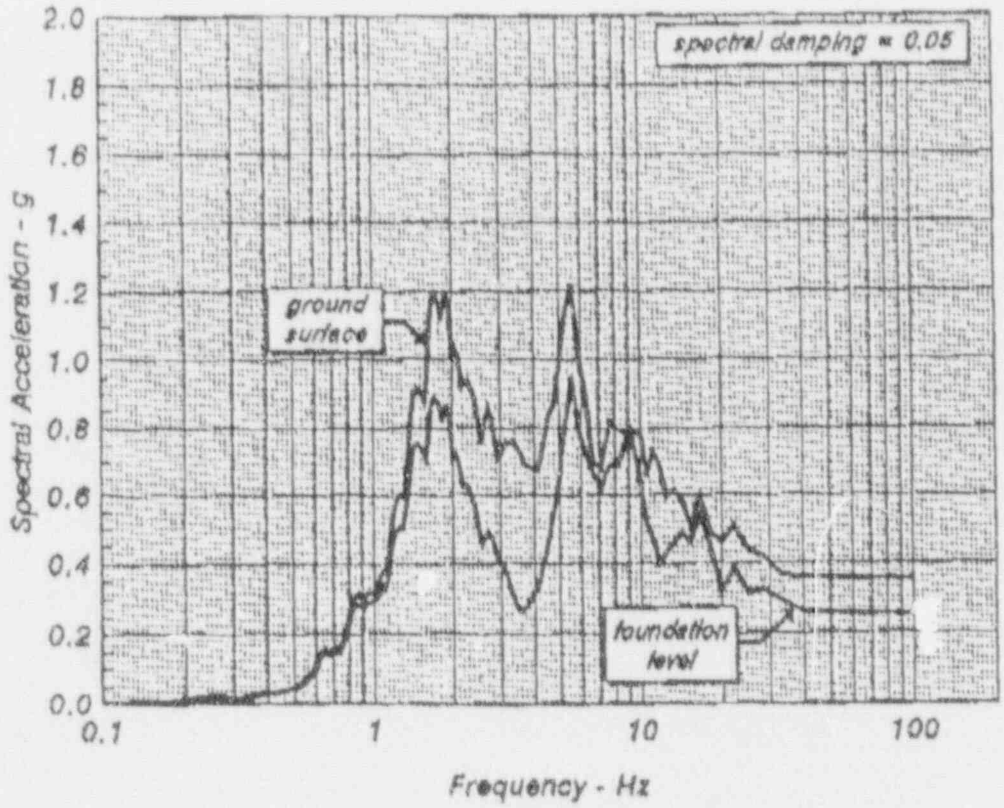


Fig. 2 Spectral Ordinates Calculated for Case B-0

IMI - 07|13|82

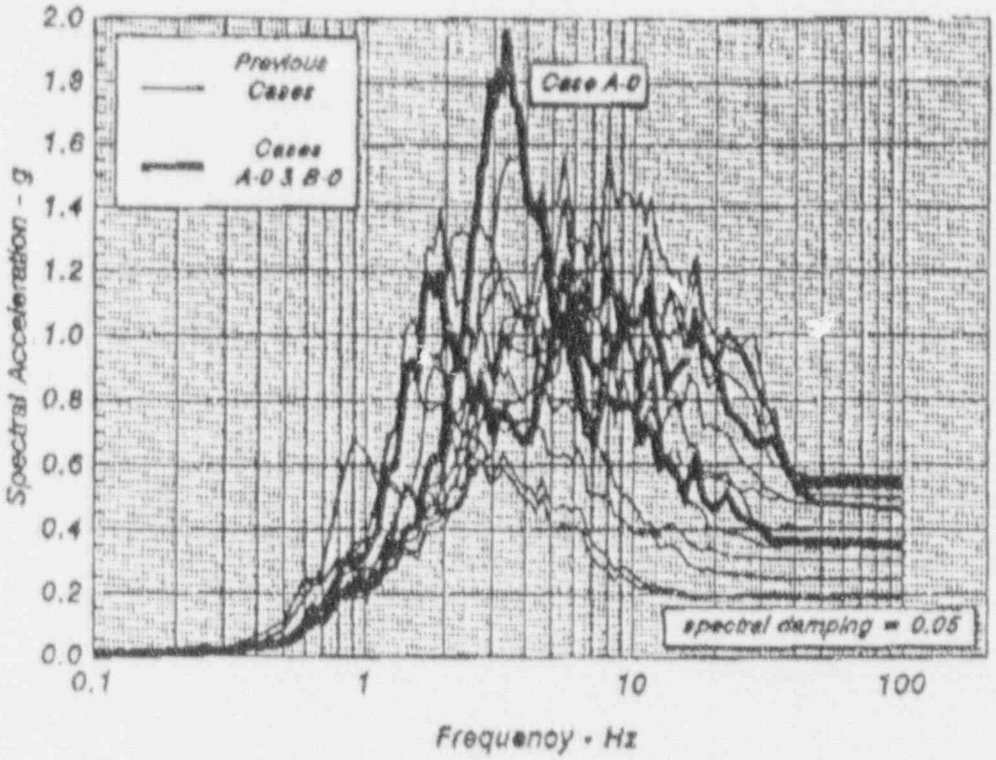


Fig. 3 Range and Average Spectral Ordinates Calculated at the Ground Surface for Previous Cases and for Cases A-0 and B-0

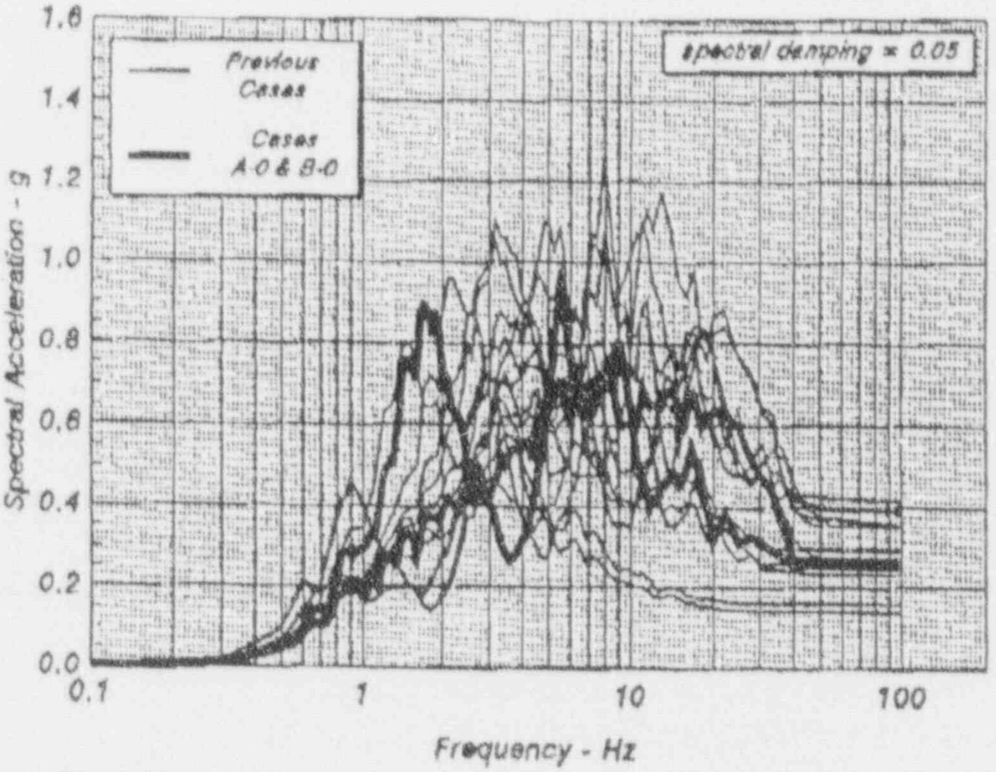


Fig. 4 Range and Average Spectral Ordinates Calculated at the Foundation Level for Previous Cases and for Cases A-0 and B-0

Question 252.09

Section I.3.a of SRP 5.2.3 requires that the RT_{NDT} of ferritic RCPB materials be determined. CE has not listed any RT_{NDT} requirements for the SA 516 Grade 70 piping in Section 5.4.3. Provide the information.

Response 252.09

Section 5.4.3.3 will be revised as shown on the attached markup.

252.09

Flow restricting orifices (7/32" dia. x 1" long) are provided in the nozzles for the RCS instrumentation and sampling lines to limit flow in the event of a break downstream of a nozzle.

5.4.3.3 **Materials**

The materials used in the fabrication of the piping are listed in Table 5.2-2. These materials are in accordance with the ASME Code, Section III. The provisions taken to control those factors that contribute to stress corrosion cracking are discussed in Section 5.2.3.

Fracture toughness of the reactor coolant piping is discussed in Section 5.2.3.

5.4.3.4 **Tests and Inspections**

Prior to, during and after fabrication of the reactor coolant piping, nondestructive tests based upon Section III of the ASME Code were performed. In addition, the fully assembled reactor coolant system is hydrostatically tested in accordance with the Code.

Inservice inspection of the reactor coolant system piping is discussed in Section 5.2.4.

5.4.4 MAIN STEAM LINE RESTRICTIONS

The steam generator outlet nozzles are one piece forgings with an integral venturi type flow restrictor. The venturi section of the nozzle is designed to reduce the flow area by 70%.

5.4.5 MAIN STEAM LINE ISOLATION SYSTEM

The Main Steam Line Isolation System (see Section 10.3.2 for more details) is composed of portions of the Main Steam System and the Engineered Safety Features Actuation System. Discussed here are those portions of these systems that respond to a Main Steam Isolation Signal, as defined in Section 7.3. A discussion of radiological considerations is provided in Section 12.3. I

5.4.5.1 **Design Bases**

A. The Main Steam Line Isolation Valves are designed to isolate the steam generators and the main steam lines in the event of a main steam line rupture.

Insert I

252.09

Insert I

The fracture toughness properties of all ferritic reactor coolant pressure boundary (RCPB) materials are required to be in accordance with the requirements of the ASME Code NB-2300 and Appendix G to 10 CFR Part 50. The SA 516 Grade 70 material used for reactor coolant piping is in accordance with these requirements.

Piping materials are required to meet the impact test requirements of NB-2300 at RT_{NDT} = 60°F or less.

Question 252.10

By letter dated October 4, 1989, CE stated that the reactor vessel fastener material conforms to the intent of R.G. 1.65 which requires that the Charpy impact energy of the reactor vessel closure studs be greater than 45 ft-lb. Provide the Charpy impact energy for the SA 540 B23 and B24 Class 3 fasteners.

Response 252.10

Section 5.3.1.7 will be revised as shown in the attached markup.

252.10

5.3.1.7 Reactor Vessel Fasteners

The bolting material for the reactor vessel closure head is fabricated from SA 540, B23 or B24, Class III material. Th material conforms to the requirements of 10 CFR 50, Appendix and the intent of Regulatory Guide 1.65, "Materials a Inspections for Reactor Vessel Closure Studs." Nondestructiv examination will be performed according to Subarticle NB-2580 oi Section III of the ASME Code, during the manufacturing process.

C-E specifies the use of a manganese phosphate coating on threads of studs, nuts and washers to improve anti-galling properties and resistance to corrosion. In addition, Super Moly lubricant (containing molybdenum disulfide) is specified to be added to threads and bearing surfaces at installation to further enhance anti-galling properties. Laboratory testing and field experience to date have shown no evidence of deleterious breakdown of either phosphate coating or lubricant.

5.3.2 PRESSURE-TEMPERATURE LIMITS

All components in the Reactor Coolant System (RCS) are designed to withstand the effects of cyclic loads due to RCS temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation.

The design number of cycles for heatup and cooldown is based on a rate of 100°F/hr. During unit startup and shutdown, the rate of temperature change is limited to less than 100°F/hr by administrative procedure.

The maximum allowable RCS pressure at the corresponding minimum allowable temperature is based upon the stress limitations for brittle fracture. These limitations are derived using linear elastic fracture mechanics (LEFM) principles, the procedures prescribed by Appendix G to Section III of the ASME Code, "Protection Against Nonductile Fracture," Appendix G to 10 CFR 50, "Fracture Toughness Requirements," NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," and the procedures recommended by WRC Bulletin 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials." Compliance with Appendix H to 10 CFR 50, "Reactor Vessel Material Surveillance Program Requirements," is discussed in Section 5.3.1.6.

Insert II

252.10

Insert II

Material for reactor vessel studs is specified to be tested in accordance with the requirements of SA 540 Grade B24, Class 3, NB 2200 and NB 2300. Charpy V-Notch testing is required to be performed at a temperature of 40°F or lower. Each specimen of one test (consisting of three specimens) is required to exhibit a minimum of 24 mils lateral expansion and 45 foot-pounds absorbed energy. These requirements also satisfy Reg. Guide 1.65 (October 1973) and Paragraph IV.A.4 of Appendix G to 10 CFR Part 50 (Nov. 20, 1979) at a temperature of 40°F or lower depending on the actual test temperature for the reactor vessel stud preload temperature or lowest service temperature, whichever is lower.

Actual material property values will be reported on the CMTR's for the reactor vessel stud material. Typical values obtained for stud materials have been evaluated from CMTR's of previously purchased materials. The results of this evaluation, which included 865 individual Charpy test data points, can be summarized as follows:

Summary of Charpy Impact Test Results

Test Temperature	Absorbed Energy (ft-lbs)			Lateral Expansion (mils)		
(°F)	Median	Mean	Std. Dev.	Median	Mean	Std. Dev.
10	52	51.3	6.9	34	33.1	4.9
40	52	51.4	3.9	32	31.4	3.5
70	57.5	56.3	10.1	40.5	40.2	7.9

These results demonstrate that the SA 540 B23 and B24, Class 3, fasteners will be able to meet the values of 45 ft-lbs. absorbed energy and 25 mils lateral expansion required for the reactor vessel closure studs.

NRC Question 281.34

The response to RAI 281.34 indicated that Section 9.1.3.3 would be revised to include part of the response. Amendment I of Section 9.1.3.3 does not include the revision.

Response to NRC Question 281.34 (Revised)

It is intended that the next amendment to CESSAR-DC will include the sampling and pool purification demineralizer resin replacement criteria stated in our previous response. This information will be included in Section 9.1.3.2.2.5 and 9.1.3.3.3 of CESSAR-DC. Revised Sections are included for information.

This response is revised to include the attached CESSAR-DC changes which were inadvertently left out of the previous response (letter LD-92-017).

ATTACHMENT 3
PFS-92-025
RAI 281.34/281.45

9.1.3.2.2.1 **Spent Fuel Pool Cooling Pumps**

Two identical pumps are installed in parallel in the spent fuel pool cooling system. Each pump is sized to deliver sufficient coolant flow through a spent fuel pool heat exchanger to meet the system cooling requirements. The pumps are horizontal, centrifugal units, with all wetted surfaces being stainless steel or an equivalent corrosion-resistant material. The pumps are controlled manually from a local station.

9.1.3.2.2.2 **Skimmer**

The skimmers are designed to circulate surface water through the spent fuel pool cleanup system and return it to the pool via the spent fuel pool cleanup pumps.

9.1.3.2.2.3 **Spent Fuel Pool Cooling Heat Exchangers**

The heat exchangers are shell-and-tube type. Spent fuel pool water circulates through the tubes while component cooling water circulates through the shell. The use of two independent heat exchangers provides redundancy so that safety functions can be performed assuming a single active failure. The tubes and other surfaces in contact with the pool water are austenitic stainless steel and the shell is carbon steel. The tubes are welded to the tubesheet to prevent leakage of pool water. I

9.1.3.2.2.4 **Pool Purification Pumps**

Two refueling pool purification pumps are used to circulate water from the refueling pool, through refueling pool demineralizers and filters. The refueling pool purification pumps are also used to circulate water from the spent fuel pool in the same fashion.

9.1.3.2.2.5 **Pool Demineralizers**

Each demineralizer is designed to provide adequate spent fuel pool or refueling pool water purity for unrestricted access to either pool working area. In addition, the demineralizers maintain pool water visual clarity.

Overtemperature protection is provided for the refueling pool demineralizers, in the event that the temperature of the spent fuel cooling water exceeds the temperature at which the ion removal capability of the resin is adversely affected.

INSERT A

ATTACHMENT 3
PFS-92-025
RAI 281.34/281.45

The design flow rate and filtering capability of the PCPS shall be such that the refueling pool water chemistry and clarity are sufficient for an operator to read fuel assembly identification numbers that are 3/8 inches high, 3/16 inches wide and 1/16 inches thick from the refueling machine at the time the operators and refueling equipment are ready to move fuel (i.e., designed such that water clarity problems do not cause refueling delays).

The PCPS shall maintain the refueling pool and spent fuel pool water chemistry and clarity within the limits specified below:

- pH between 4.5 and 10 @ 25°C;
- Chlorides less than 0.15 ppm; and
- Optical clarity consistent with the requirements as described above

INSERT B

9.1.3.4 Tests and Inspections

Components of the PCPS may be in either continuous or intermittent use during normal system operation. Periodic visual inspection and preventive maintenance are conducted using normal industry practice. The Seismic Category I portions will be inspected in accordance with the ASME B&PV Code, Section XI. I

No special equipment tests are required since system components are normally in operation when spent fuel is stored in the fuel pool.

Sampling of the fuel pool water is performed for gross activity and particulate matter concentration. The layout of the components of the PCPS is such that periodic testing and inservice inspection of this system are possible.

9.1.3.5 Instrumentation Application

The instrumentation provided for the PCPS is discussed in the following paragraphs. Alarms and indications are provided as noted.

INSERT A

Demineralizer resin replacement is to be based on three criteria:

- Breakthrough of cesium, cobalt, chloride, or fluoride.
- Pressure drop not to exceed demineralizer and resin vendors' recommended limit for the as procured equipment.
- Thermal excursion approaching the resin vendors' recommended limit for the as procured equipment.

INSERT B

The refueling pool and spent fuel pool will be monitored via grab samples to ensure the water quality is maintained within these limits. The PCPS demineralizer effluent will also be monitored by grab sample in order to determine when resin breakthrough has occurred.

NRC Question 281.45

The spent fuel pool demineralizer replacement criteria is not included in Amendment I as is indicated in applicant's response to NRC Question 281.34.

Response to NRC Question 281.45

Please refer to the response to Question 281.34.

Question 252-13: USI A-49 Pressurized Thermal Shock (PTS)

The staff has revised equations in 10 CFR Part 50.61 that calculate the limiting reference temperature RTpts. The revision was published in the Federal Register, Volume 56, Page 22300, May 15, 1991. The revised equation will change the RTpts of 109 F that CE has calculated. CE needs to recalculate the RTpts and to revise the corresponding sections in CESSAR-DC.

Response 252-13: (Revision 1)

The updated RTpts has been calculated according to the revision of 10 CFR Part 50.61 as specified in the Federal Register, Volume 56, Page 22300, May 15, 1991.

The new value for RTpts is 97 degrees F. Applicable sections of CESSAR-DC will be revised in a future amendment to incorporate this change.

Attachment (2) to PFS-92 016
Page 3 of 6

252.13

The design relief capacity of each of two SCS relief valves (shown in P&ID Figure 6.3.2-1B) as supplied by the valve manufacturer meets the minimum required relief capacity of 4000 gpm which contains sufficient margin in relieving capacity for even the worst transient. The SCS relief valves are Safety Class 2, designed to Section III of the ASME Code. E

5.2.2.10.2.4 Administrative Controls

Administrative controls necessary to implement the LTOP provisions are limited to those controls necessary to open the SCS isolation valves.

During cooldown, when the temperature of the RCS is above that corresponding to the intersection of the controlling P-T Limit and the pressurizer safety valve setpoint, overpressure protection is provided by the pressurizer safety valves, and no administrative procedural controls are necessary. Before entering the low temperature region for which LTOP is necessary, RCS pressure is decreased to below the maximum pressure required for LTOP. The LTOP pressure is less than the maximum pressure allowable for SCS operation. Once the SCS is aligned, no further specific administrative procedural controls are needed to ensure proper overpressure protection. The SCS will remain aligned whenever the RCS is at low temperatures and the reactor vessel head is secured or until an adequate vent has been established. As designated in Table 7.5-2, indication of SCS isolation valve position is provided. E I

During heatup, the SCS isolation valves remain open at least until the LTOP enable temperature. Once the RCS temperature has reached that temperature corresponding to the intersection of the controlling P-T Limit and the pressurizer safety valve setpoint, overpressure protection is provided by the pressurizer safety valves. The SCS can be isolated and no further administrative procedural controls are necessary. E

5.2.2.11 Pressurized Thermal Shock

The System 80+ reactor vessel meets the requirements of 10 CFR 50.61, "Fracture Toughness Requirements For Protection Against Pressurized Thermal Shock Events." The calculated RT_{PTS} is 109°F 97°F, which satisfies the screening criteria in 10 CFR 50.61(b)(2).

* Insert A

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

5.2.3.1 Material Specification

A list of specifications for the principal ferritic materials, austenitic stainless steels, bolting and weld materials, which are part of the reactor coolant pressure boundary is given in Table 5.2-2.

252.13

Insert A:

This number has been calculated with the following assumptions:

1. The maximum initial RT_{NDT} for the vessel beltline will be +10°F. It will, however, be a goal to achieve -20°F initial RT_{NDT} for the reactor vessel beltline materials.

2. The adjustment in the reference temperature caused by irradiation (ΔRT_{PTS}) is +53°F. This calculated value assumes a forging with 0.06 wt-% maximum copper content, and a 1.00 wt-% maximum nickel content.

3. The margin added for uncertainties is +34°F.

252.13

These material and fabrication techniques and other reactor vessel design features are described as follows:

- The copper content is controlled to assure that the RT_{PTS} will remain acceptable over the life of ~~the plant~~.

- The characterization of the steel and ~~weel~~ weld materials was established through industrial and governmental studies which examined the material properties in both the unirradiated and the irradiated condition. Inservice inspection and material surveillance programs are also conducted during the service life of the vessel, further ensuring adequate vessel integrity and safety margin.

- Design, materials of construction, fabrication methods, inspection requirements, shipment and installation, operating conditions, and inservice surveillance are all components of a program to assure reactor vessel integrity for the plant design lifetime. A complete description of the reactor vessel design is given in CESSAR-DC, Section 5.3.

- The System 80+ Standard Design reactor vessel is fabricated from ring forgings, thus eliminating vertical welds in the beltline region where neutron irradiation is greatest. The elimination of these particular welds further reduces the possibilities of impurities in weld material which are known to result in an RT_{PTS} that approaches the screening criterion of 270 degrees F.

- Furthermore, the System 80+ Standard Design reactor vessel meets the requirements of 10 CFR 50.61 as described in CESSAR-DC, Section 5.2.2.11. Specifically, the calculated RT_{PTS} at the end of the 60-year service life is 109 97 degrees F, which is significantly below the screening criterion of 270 degrees F for plates, forgings and axial weld materials, or 300 degrees F for circumferential weld materials.

Since the System 80+ reactor vessel design complies with the ASME code and other accepted industry codes and standards, and meets the requirements of 10 CFR 50.61, this issue is resolved for the System 80+ Standard Design.

REFERENCES

1. NUREG-0933, "A Status Report on Unresolved Safety Issues", U. S. Nuclear Regulatory Commission, April 1989.

Question 4 20.62

In the response to Question 420.32, C-E defined a proven product as equipment or commercial software which has been in the field for at least 3000 operating years or has an equivalent installed base. Provide the basis for this value.

Response 4 20.62

The "3000 operating years" value stated in C-E's previous response is based on the EPRI ALWR requirement for a proven product, which is three years of operation. It is our opinion that the requirement of 3000 operating years reflects feedback from a large base of operating products, in addition to an adequate length of time in service.

The C-E value is expressed in operating years in order to provide specific guidance to our design engineers. The C-E value would require 1000 components in operation for three years, or 3000 components in operation for one year, to be proven. C-E believes that one year is the minimum acceptable in-service interval regardless of quantity.

The 3000 operating years value was established based on C-E's engineering judgement to define a baseline for a mature, proven, commercial product. This value is not a firm requirement, but rather a benchmark for products that would require further justification. It can be expected that some limited applications may require commercial products with less than 3000 operating years experience. For these cases either additional justification, which establishes the proven status of a product based on similarity to a proven product is required, or additional component burn-in and factory testing would be required.

Question 420.63

In the response to Question 420.45, C-E states that the responses to Questions 420.23 and 420.33 describe C-E's V&V activities, QA audit process, testing, and log of all software problems, and software configuration management, and that these activities can be relied upon to maintain a virus-free environment. Because neither reference adequately describes antivirus activities, please supplement your response.

Response 420.63

Although the references do not specifically identify antivirus activities, the methodologies as a whole represent practices that have resulted in virus free environments. C-E has successfully produced virus free software in safety systems for 15 years following its V&V, QA and configuration management practices. CEN-39, CPC Protection Algorithm Software Change Procedure, previously accepted by the NRC, provides additional details on the software process.

Question 420.64

In response to Question 420.54, C-E states that they know of no negative aspects associated with the use of digital technology in nuclear power plants. The incident at Bruce Nuclear Generating Station in Ontario, Canada and the Electricite de France (EDF) decision to abandon the P20 Controlbloc design in the French N4 plants, are two examples of the negative aspects associated with digital technology in the nuclear industry. Since there is limited use of digital equipment in the nuclear industry, the staff requests that C-E expand the scope of their research to better understand the types of system failures that can occur in a digital system. Provide the scope of the studies performed to date regarding digital systems implications in safety critical applications.

Response 420.64

In its Policy Issue SECY-91-292 on Digital Computer Systems for Advanced Light Water Reactors, September 16, 1991, the NRC concludes that "There is a general consensus within the international nuclear community that the proper use of digital computer technology in the design of monitoring, control and protection systems will improve the safety and performance of nuclear power plants."

C-E agrees with that conclusion and in its response to Question 420.54 cited some of the advantages of digital technology which support that conclusion.

The key words in the NRC conclusion are "proper use." Digital technology in itself has no negative aspects. If one considers the potential of improper use as a negative aspect, all technology shares that same potential for improper use, and thus the same negative aspect.

C-E recognizes that the very attributes of digital technology which give rise to its potential for improved safety and performance (i.e. software based control, shared data, communications, processing power, etc.) also raise valid concerns with respect to design reliability, particularly in the area of software, and give emphasis to the need for a structured design process and the use of formal verification and validation quality procedures.

C-E is aware of the incidents cited by the NRC as examples of negative aspects associated with digital technology and considers those as examples of design problems, not digital technology problems.

The incident at the Bruce Nuclear Generating Station has been attributed by the Atomic Energy Control Board (AECB) to a long standing error in the computer software controlling the refueling machines which surfaced under a specific set of conditions. The problem with the P20 Controlbloc design in the French N4 plants is tied to the revolutionary nature of the design and its reliance on the development of new, unproven product technology.

420.64

By contrast, the Nuplex 80+ design is an evolutionary design employing proven design principles and standard, commercially available, field proven products. This conservative approach employed in the Nuplex 80+ design is an example of the proper use of digital technology to improve plant safety and performance.

C-E continuously monitors industry activities related to the reliability of digital based control and protection systems. When incidents such as those cited occur, C-E closely monitors the events through direct industry contacts and published articles to determine the applicability to the Nuplex 80+ design. Appropriate changes, based on lessons learned from these incidents, are incorporated into the design and into the design process to continuously improve the reliability of the system.

Recent activities conducted by C-E related to the implications of the use of digital technology in safety critical applications include:

- An assessment of the design problems in the French P20 Controlbloc design was conducted.

- Direct contacts with Atomic Energy of Canada Limited (AECL) were made to share information related to the use of digital computer based commercial products in safety applications and to review AECL's status relative to design and licensing activities and problems.

- A review of the software design and licensing progress for Sizewell B as it relates to the British Nuclear Fuels Limited Feasibility Study was conducted through C-E's contacts with AEA Technology.

Question 430.9 (8.3.1)

One of the major differences between the distribution systems specified in CESSAR-DC Chapter 8 and those found in most recently licensed nuclear plants is that no alternate power source is provided for the non-safety loads required for unit operation (Figure 8.3.1-1). Since only the safety and permanent non-safety loads have a transfer capability to the reserve source of offsite power, the non-safety loads such as the reactor coolant pumps, feedwater pumps, condensate pumps, and circulating water pumps do not have an alternate source of power. As a result, a loss of power to these loads which could be caused by a failure (fault) anywhere on the Unit Auxiliary Transformers, Main Step-Up Transformers or their connecting feeders results in a plant trip and loss of RCS forced circulation and normal feedwater systems. Reliance must therefore be placed on natural circulation and the safety systems. The CESSAR-DC proposed design therefore results in increased reliance on the challenges to the safety systems, and perhaps more frequent excursions towards peak clad temperatures. We therefore recommend that you evaluate alternatives that would provide a second source of power to these non-safety loads.

Response 430.9

Figure 8.3.1-1 has been revised to show the safety buses being fed directly from offsite power (See RAI 430.13). Safety and non-safety buses now have a normal and an alternate power supply which are independently tied to transformers in the switchyard. Chapter 8 text has been revised to reflect the new electrical one line.

See revised CESSAR-DC sections from chapter 8.

8.0 ELECTRIC POWER

8.1 INTRODUCTION

An offsite power system and an onsite power system are provided to supply the unit auxiliaries during normal operation and the Reactor Protection System and Engineered Safety Feature Systems during abnormal and accident conditions. E

8.1.1 OFFSITE POWER SYSTEM

The typical offsite power transmission system grid may consist of interconnected hydro plants, fossil-fueled plants, combustion turbine units, and nuclear plants supplying energy to the service area at various voltages. I

The unit is connected to a switchyard and ~~thereby~~ to the transmission system via two separate and independent transmission lines. The generator circuit breakers, along with the unit step-up transformers, allow ~~one of~~ these lines not only to supply power to the transmission system during normal operation, but also to serve as an immediately available source of preferred power. Insert Y → ~~The other separate transmission line is connected, via the switchyard and a standby auxiliary transformer, to provide an independent second immediate source of offsite power to the onsite power distribution system for safety and permanent non-safety loads~~. E main

A description of a representative offsite power system is provided in Section 8.2.

8.1.2 ONSITE POWER SYSTEMS

The onsite power system for the unit, as two depicted on Figure 8.1-1, consists of the main generator, ~~the~~ generator circuit breakers, unit main transformers, the unit auxiliary transformers, safety systems → ~~standby auxiliary~~ transformers, the diesel generators, an alternate AC source, the batteries, and the auxiliary power system. Under normal operating conditions, the main generator supplies power through isolated phase bus and generator circuit breakers to the unit main step-up and unit auxiliary transformers. The unit auxiliary transformers are connected to the bus between the generator circuit breakers and the unit main transformers. During normal operation, station auxiliary power is supplied from the main generator through these unit auxiliary transformers. During startup and shutdown, the generator circuit breakers ~~is~~ are open, and station auxiliary power is supplied from the transmission system through the unit main and unit auxiliary power transformers. I E

RAI 430.9

INSERT Y

"A third separate and independent transmission line is connected to a switchyard containing two circuit breakers and safety system transformers, as illustrated by Figure 8.2-2, to provide another source of offsite power to the 4160V safety buses."

8

The Class 1E safety loads are divided into two redundant and independent load group Divisions I and II. Each Load Division is capable of being supplied power from the following sources, listed in decreasing order of priority:

A. Safety System Transformers

B ~~A.~~ Unit Main Turbine Generator

C ~~B.~~ Unit Main Transformers ~~(Offsite Preferred Bus-1)~~

~~C. Standby Auxiliary Transformer (Offsite Preferred Bus-2)~~

D. Emergency Diesel Generators

E. Alternate AC Source

Priorities A-C may be reversed depending on plant specific offsite power reliabilities.

If both the offsite power sources and the standby emergency diesel generators are unavailable, either one of the Divisions may be powered independently from the Alternate AC (AAC) Source. The AAC is a Non-Class 1E gas turbine which provides an independent and diverse power source. The AAC source is furnished with a battery and charger to provide power to the associated DC loads. E I

A 125V DC Vital Instrumentation and Control Power System is available to provide power to the Class 1E DC loads and the diesel generators. Additionally, this system provides power to Class 1E 120V AC loads through inverters. E

The unit also has a 125V DC Auxiliary Control Power System and a 250V DC Auxiliary Power System to supply essential Non-Class 1E DC loads. Additionally, this system also provides power to Non-Class 1E 208/120V AC loads through inverters.

The onsite power systems are described in detail in Section 8.3.

8.1.3 DESIGN BASES

The design bases for the offsite power system and the onsite power system are presented below.

A. Offsite Power System

Insert C → 1. ~~Each of the two offsite power circuits has sufficient capacity, is normally energized, and is available to supply power to the plant safety-related systems within a few seconds following a loss-of-coolant accident (LOCA) to assure that core cooling, containment integrity, and other vital safety functions are maintained.~~ I E

RAI 430.9

INSERT C

" 1. Each one of the three offsite power circuits is normally energized, and have sufficient capacity to adequately power the plant safety related systems following a loss-of-coolant accident (LOCA) to assure that core cooling, containment integrity, and other vital safety functions are maintained. The circuit to which the safety buses are normally connected is available to supply power to the safety loads within a few seconds following a LOCA. The other two offsite power circuits are available to provide power to the safety loads within a time shown to be acceptable by the plant design bases."

2. The ~~two~~ three offsite power circuits (to the switchyard) are designed to be independent and physically separate to assure their availability under normal and postulated accident conditions.

B. Onsite Power System

1. The Class 1E onsite power systems are located in Seismic Category I structures to provide protection from natural phenomena.

2. The redundant Class 1E onsite power system equipments are located in separate rooms or fire zones with adequate independence to assure that the Plant Protection System safety functions can be performed assuming a single failure.

3. Voltage levels at the Class 1E safety-related buses are optimized for the full load and minimum load conditions that are expected throughout the anticipated range of voltage variations of the power source by the adjustments of the voltage tap settings on the transformers.

4. The Class 1E onsite power systems have sufficient capacity to safely shut the unit down and to mitigate the effects of an accident assuming loss of offsite power (LOOP).

5. The Class 1E onsite power systems are designed to permit appropriate surveillance, periodic inspections, and testing of important areas and features to assess the continuity of the systems and the condition of their components. E

6. The emergency diesel generators are designed to be automatically initiated in the event of an accident or a LOOP. I

7. The vital batteries have adequate capacity, without chargers, to provide the necessary DC power to perform the required safety functions in the event of a postulated accident assuming a single failure. E

8. Each vital battery charger has adequate capacity to supply its assigned steady-state loads while simultaneously recharging its associated battery.

9. A non-Class 1E AAC source is provided to help mitigate the effects of LOOP and station blackout (SBO) scenarios.

Y. Regulatory Guide 1.118

The periodic testing requirements of the electric power and protection system are presented in Chapter 16, Technical Specifications.

Z. Regulatory Guide 1.128

The installation design and installation of Class 1E batteries are in compliance with the intent of Regulatory Guide 1.128 as discussed in Section 8.3.2.1.2.1.

AA. Regulatory Guide 1.129

Maintenance, testing, and replacement of large lead batteries complies with the intent of Regulatory Guide 1.129. E

BB. Regulatory Guide 1.131

The qualification testing of electric cables, field splices, and connections complies with the intent of Regulatory Guide 1.131.

CC. Regulatory Guide 1.155

permanent

The installation and design of the onsite AAC power source system is in compliance with the intent of Regulatory Guide 1.155 for a station blackout (SBO). The AAC power source is designed to be made available to power one safety load division and its corresponding ~~essential~~ non-safety load bus within 10 minutes of the onset of the SBO; such that the plant is capable of maintaining core cooling and containment integrity per Section 50.63 of 10 CFR Part 50.

The AAC source is not normally directly connected to the plant's main or standby offsite power sources or to the Class 1E Safety Division power distribution system. There is a minimum potential for common cause failure with the offsite power system or with the emergency diesel generators.

The AAC power source is further discussed in Section 8.3.1.1.5. I

DD. Regulatory Guide 1.158

The qualification testing of safety-related lead storage batteries complies with the intent of Regulatory Guide 1.158.

8.1.4.3 IEEE Standards

A. IEEE Standard 387-1984

The preoperational and periodic testing of the emergency diesel generators complies with the requirements of IEEE Standard 387-1984 as discussed in Section 8.3.1.1.4.11.

B. IEEE Standard 741-1986

Protection for degraded voltage and loss of voltage conditions for safety and non-safety buses is provided, as described in Section 8.3.1.1.6.

C. IEEE Standard 765-1983

three

The offsite preferred power supply and its interface with the onsite power system comply with IEEE Standard 765-1983. The offsite supply consists of ~~two~~ independent transmission lines as discussed in Sections 8.2.1.3 and 8.2.1.4. These transmission lines are designed to minimize the probability of their simultaneous loss due to ~~a pylon~~ failure or a failure ~~of a~~ crossings transmission lines. The switchyard design minimizes the probability of a single equipment failure causing the simultaneous loss of both preferred power supply circuits. E

due to other tower

I

8.2 OFFSITE POWER SYSTEM

8.2.1 SYSTEM DESCRIPTIONS — E

8.2.1.1 Utility Grid System

The utility grid system, which is not within the scope of the System 80+ Standard Design, may consist of interconnected hydro, fossil fueled and nuclear plants supplying energy to the service area at various voltages. The grid transmission system is also a source of reliable and stable power for the onsite power distribution system. The grid system design ~~must~~ SHALL include at least ~~two~~ three preferred power circuits, each capable of supplying the plants' necessary safety loads and other equipment. — I, E

8.2.1.2 Utility Grid and Switchyard Interconnections

The switchyards ~~is~~ are connected to the primary transmission system by overhead transmission lines. Figures 8.2-1 and 8.2-2 depict ~~a~~ typical interconnections of the switchyards and onsite power. Transformers, fed from Preferred Switchyard Interface I, which shall be

8.2.1.3 Station Switchyard

Two ~~T~~transmission lines from the primary transmission system shall terminate in the preferred Interface I switchyard with provisions for additional lines to be added in the future. ~~Additionally,~~ the Unit and ~~Standby Auxiliary Transformers~~ Safety Systems Transformers, fed from Preferred Switchyard Interface II, which shall be are tied to the switchyards by separate and independent overhead lines.

The entire Preferred switchyard Interface I, including the power circuit breakers, cabling system, AC and DC auxiliary power systems, protective relaying system, and control system shall be divided into two preferred power buses designated 1 and 2. These designations shall be consistent with the preferred power feeder designations. Additionally, the incoming transmission lines shall be also assigned to power buses in such a way as to separate the associated cabling, protective relaying, and controls for each circuit transmission line into two distinct sources of offsite power.

The Preferred switchyard Interface I design shall provide redundant offsite power feed capability to the nuclear unit.

8.2.1.3.1 Switchyard 480V AC Auxiliary Power System

A 480V AC Auxiliary Power System shall be provided in the switchyard to supply a reliable source of continuous AC power for the power circuit breaker auxiliaries, battery chargers, relay house air conditioning, and switchyard lighting.

— The Unit Main and Safety Systems Transformers ~~are~~ shall be physically separated such that no fire nor environmental effect shall disable both the safety and non safety offsite sources.

The status of this system shall be monitored in the switchyard relay house with annunciators and in the control room via the Data Processing System (DPS) and Discrete Indication and Alarm System (DIAS) systems described in Chapters 7 and 18.

8.2.1.3.2 Switchyard 125V DC Auxiliary Power System

A 125V DC Auxiliary Power System shall be provided to supply a reliable source of continuous DC power for all relaying, control, and monitoring equipment in the switchyard. This system shall consist of two independent ~~trains~~ redundant divisions each supplying DC power to its associated preferred power bus equipment.

8.2.1.3.3 Switchyard Protective Relaying System

The Switchyard Protective Relaying System shall be provided to protect switchyard equipment and to contribute to power system stability by promptly and reliably removing a transmission line and/or switchyard bus from service under a fault or an abnormal condition.

8.2.1.3.4 Switchyard Control System

The Switchyard Control System shall consist of all control circuits for operating switchyard power circuit breakers (PCBs) and motor-operated disconnect switches (MODs). Controls shall be provided, via the Process-CCs described in Section 7.7, in the main control room for the PCBs and MODs associated with the unit feeders.

E

In addition to the controls provided in the main control room, each PCB or MOD shall be able to be operated at the switchyard relay house or at the local control cabinet of the PCB or MOD.

8.2.1.4 Switchyard and Station Interconnections

Two separate and physically independent overhead transmission line circuits are provided to connect the preferred switchyard Interface I to the Unit Main Transformers. These transmission lines shall be designed to withstand the heavy loading conditions defined in the National Electric Safety Code.

Compliance with General Design Criterion 17

The offsite power system is designed with sufficient independence, capacity, and capability to meet the requirements of GDC 17. The transmission network is connected to the onsite power system by ~~two~~ three physically independent circuits.

A third separate and physically independent overhead transmission ~~line~~ circuit is provided to connect the Preferred Switchyard Interface II to the safety systems transformers.

The offsite power system shall be designed to minimize the probability of losing electric power from any supplies as a result of or coincident with the loss of the unit generator, the transmission network, or the onsite electric power supplies.

Compliance with General Design Criterion 18

The requirements of General Design Criterion 18 shall be implemented in the design of the offsite power system. The design shall permit periodic inspection and testing of important areas and features. The design shall include the capability to periodically test the operability and functional performance of the components of the systems as a whole and under conditions as close to design as practical.

8.2.1.5 Offsite Power System Operational Description

The nuclear generating unit shall be provided with two independent immediate access circuits of offsite power. Prior to and during startup of the nuclear unit, the Unit Auxiliary Power System shall receive power from the transmission system through the main unit main transformers and the unit auxiliary transformers. During this period, the generator circuit breakers and associated disconnect switches shall be open. (The safety bus shall receive power from the transmission system through the Safety System Transformers.) After the unit generator has been brought to rated speed and its field applied, the unit generator shall be then connected to the system by closing the generator circuit breakers. Automatic and manual synchronization are provided and supervised by synchronizing relays. E

8.2.1.5.1 Offsite Power System Protective Relaying

The offsite power system protective relaying system shall be designed to remove from service with precision and accuracy any element of the offsite power system subjected to an abnormal condition that may prove detrimental to the effective operation or integrity of the unit.

8.2.1.6 Reliability Considerations

The transmission system shall be designed to conform to the reliability criteria established by the owners appropriate electric reliability council. Typically, transmission systems are designed to avoid system cascading upon the occurrence of any one of the following:

A. Loss of Generation

1. Sudden loss of entire generating capability in any one plant.

TABLE 8.2-1

(Sheet 1 of 2)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE OFFSITE POWER SYSTEM

Component	Malfunction	Resulting Consequences	
1. Transmission System — Preferred Switchyard Interface I	Loss of Power	(a) The switchyard PCBs connecting the unit to the system (switchyard) trip automatically.	E
		(b) If main turbine generator is available, ~~the emergency diesel generators and AAC source start~~. All unit ~~and~~ non Class 1E auxiliaries continue to receive an uninterrupted flow of power from the main turbine generator through the main generator circuit breaker.	I E I
		(c) If the main turbine generator is not available, the ~~emergency diesel generators and~~ AAC source starts and the loads are sequenced on automatically to provide power to their respective ~~Class 1E and~~ permanent non-safety loads.	E I E
2. Redundant switchyard bus - Preferred Switchyard Interface I	Loss of one	(a) No consequence. The redundant PCBs (as applicable) trip. The unit is still connected to the system through the remaining switchyard bus.	
3. Preferred Switchyard Interface I power circuit breakers connecting the step-up transformers to the switchyard	Loss of one due to a ~~fault~~ breaker failure	(a) The faulted equipment is isolated by protective relaying and protective equipment. (b) The other independent offsite circuit remains unaffected.	

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"4.	Transmission System-Preferred Switchyard Interface II	Loss of power	(a) A loss of offsite power is experienced on the 4160V safety buses. (b) The diesel generators automatically start. (c) Loads are sequenced on to the diesel generator. (d) If offsite power is available from Preferred Switchyard Interface I, the diesel generator is manuallysynchronized with the permanent non-safety switchgear. 4160V loads are then fed from power from the permanent non-safety switchgear."

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"5. Preferred Switchyard Interface II power circuit breakers connecting step-up transformers to the switchyard.	Loss of one due to breaker failure.	(a) The faulted equipment is isolated by protective relaying and protectiveequipment. (b) A loss of offsite power is experienced on the 4160 V safety buses. (c) The diesel generators automatically start. (d) Loads are sequenced on to the diesel generator. (e) If offsite power is available from Preferred Switchyard Interface I, the diesel generator is manuallysynchronized with the permanent non-safety switchgear. 4160V loads are then fed from power from the permanent non-safety switchgear."

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Attachment ALWR-376

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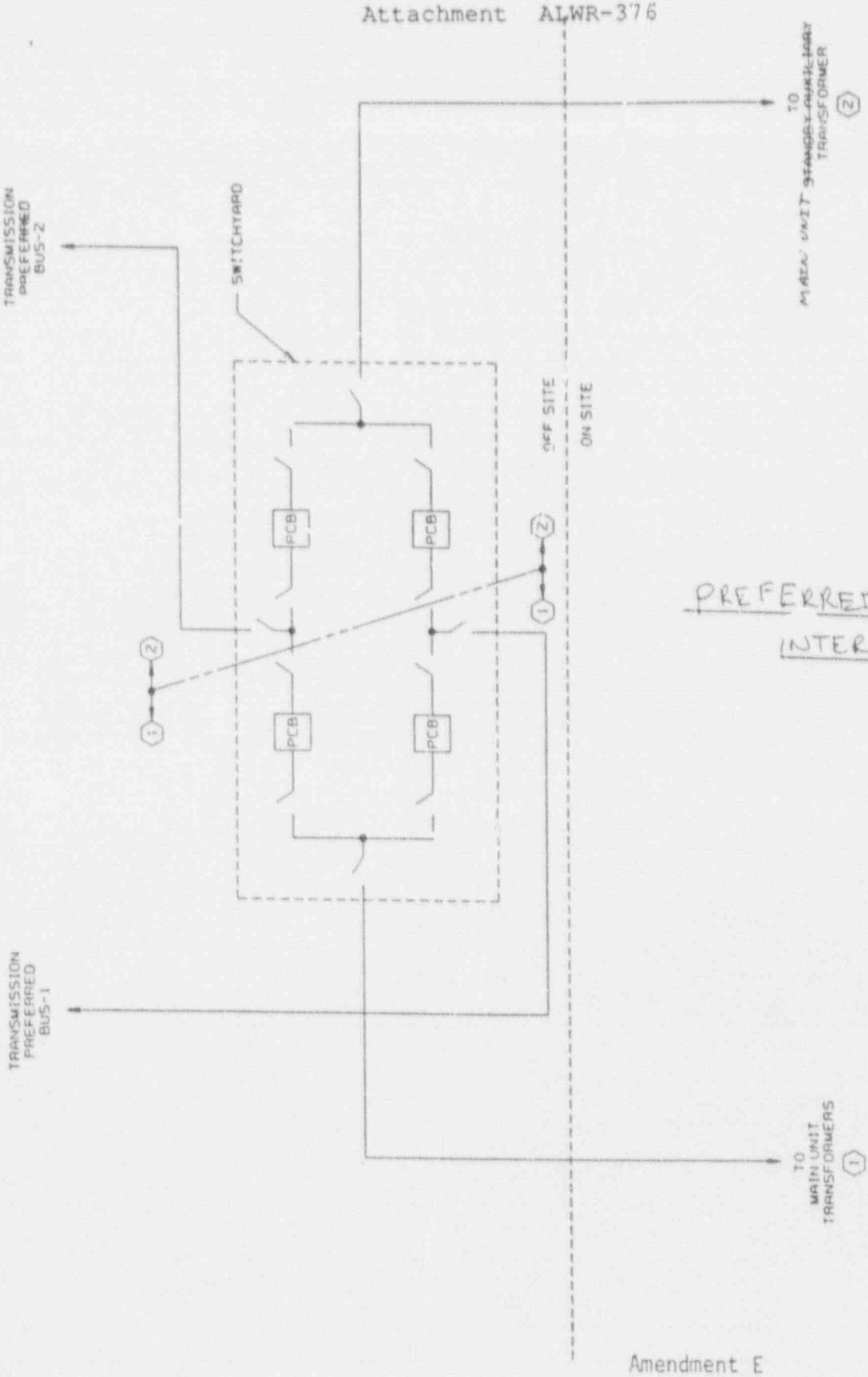

PREFERRED SWITCHYARD INTERFACE I

LEGEND:

= MOTOR OPERATED DISCONNECT SWITCH

PCB = POWER CIRCUIT BREAKER

(X) = POWER BUS DESIGNATION 1,2

Amendment E
December 30, 1988

SYSTEM 80+™	TYPICAL ~~ONSITE~~ POWER SYSTEM INTERCONNECTION WITH OFFSITE POWER	Figure 8.2-1

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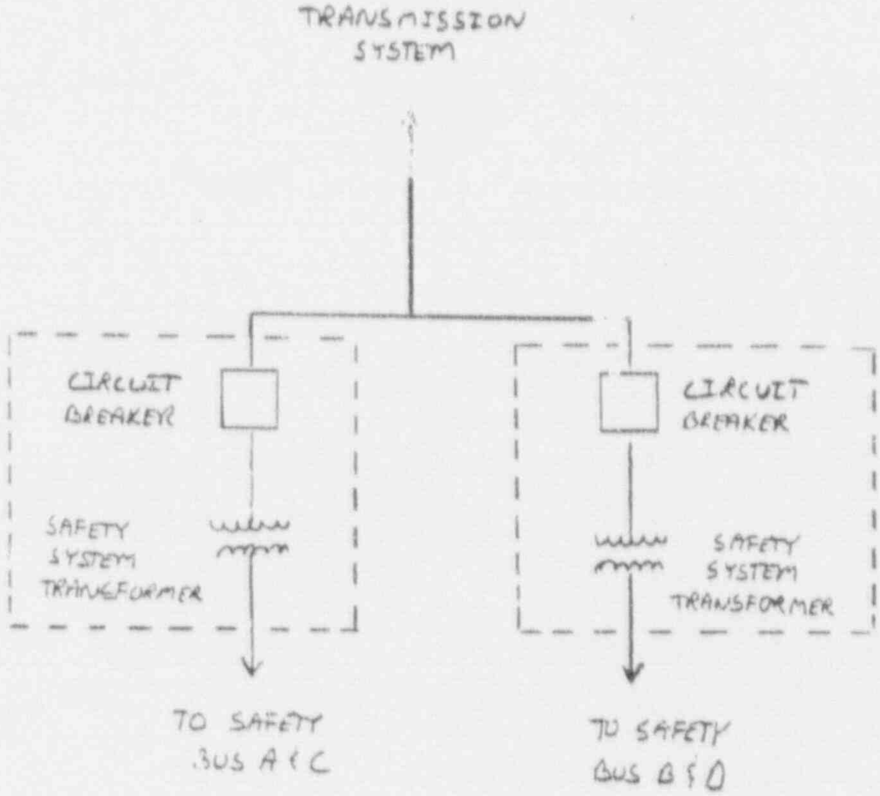

PREFERRED SWITCHYARD INTERFACE II

TYPICAL POWER SYSTEM INTERCONNECTION WITH OFFSITE POWER

FIGURE 8.2-2

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8.3 ONSITE POWER SYSTEMS

8.3.1 AC POWER SYSTEMS

8.3.1.1 System Descriptions

8.3.1.1.1 Non-Class 1E AC Power Systems

8.3.1.1.1.1 Unit Main Power System

The Unit Main Power System, as shown on Figure 8.3.1-1, consists of the unit main turbine generator, two associated isolated phase bus, four generator circuit breakers, six single-phase ~~four~~ unit main transformers with one spare and four ~~two~~ 50% capacity unit auxiliary transformers. The primary function of this system is to generate and transmit power to the transmission system while simultaneously supplying power to the unit auxiliaries. In the event that the main generator is not in service, this system is used to supply power from the transmission system to the unit auxiliaries. The design bases for the Unit Main Power System are discussed in Section 8.1.3. E

8.3.1.1.1.2 13,800 Volt Normal Auxiliary Power System

The 13,800V Normal Auxiliary Power System consists of four ~~two~~ non-safety switchgear. ~~groups. The first switchgear group designated "X" is normally connected through a main breaker to one of the dual voltage 13,800/4,160V Unit Auxiliary Transformers. A second switchgear group designated "Y" is normally connected to the other independent Unit Auxiliary Transformer.~~ INSERT A

The 13,800 Volt Normal Auxiliary Power System furnishes power to large motors such as the reactor coolant pump motors and condensate water pump motors.

The protective relaying for the 13,800V switchgear feeders and buses can be classified as follows:

A. Protection of large motors.

B. Protection of buses and bus feeders.

The protective schemes are designed to isolate the faulted equipment from the rest of the system, to minimize the effect of the fault and to maximize availability of the remaining equipment. The scheme also limits the damage and the time out of service of the faulty equipment. Each scheme is designed to best achieve this for the specific equipment protected. The basic schemes consist of ground fault protection, instantaneous overcurrent and timed overcurrent protection. Other forms of

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"Each switchgear is normally separately tied to its own secondary winding of a dual low voltage 13,800/4,160V Unit Auxiliary Transformer through its normal incoming breaker. Should the switchgear lose its normal source, it will automatically transfer to its alternate incoming breaker to share another switchgear's Unit Auxiliary Transformer. If the the residual voltage of the 13.8KV motors are in synchronism with the alternate source, a fast transfer will result. If they are out of synchronism, the transfer will be delayed until the residual voltage is 25% or less."

protection are provided where applicable and consist of current or undervoltage differential and reverse power flow protection. Each breaker in this auxiliary power system is provided with timed overcurrent protection and an anti-pump device.

Each switchgear assembly has a short circuit capability which is verified by manufacturers prototype tests and exceeds the short circuit requirements of the 13,800V Normal Auxiliary Power System.

8.3.1.1.1.3 4,160 Volt Normal Auxiliary Power System

The 4,160V Normal Auxiliary Power System consists of four switchgear groups and a non-Class 1E Alternate AC source. The first switchgear group designated "X" is connected to the Unit Auxiliary "X" Transformer to power large non-safety loads such as Station service water system ~~main circulating water~~ pumps, turbine building Secondary ~~Circulating~~ Water System ~~CCW~~ pumps, etc. The second switchgear group "Y" is connected to the other Unit Auxiliary "Y" Transformer to power the similar ~~remaining large~~ non-safety loads.

The third switchgear group designated Permanent Non-safety "X" provides power to auxiliary and service loads which must typically remain operational independent of the plant operating conditions or during plant outages (such as CVCS charging pump and building supply fans). Its normal source is preferred power from the ~~4,160V~~ Unit Auxiliary Transformer "X". In the event that its normal source is lost, this switchgear may be connected ~~either to the~~ Unit ~~Standby~~ Auxiliary Transformer ~~(preferred bus 2)~~ "Y" or to the Alternate AC source. An interlock is provided between the normal (preferred-1) and alternate ~~standby~~ (preferred-2) power source breakers to preclude them from both being closed simultaneously. E

~~In case of failure of the normal power source, i.e., the Unit Auxiliary Transformers, without loss of offsite power; the Permanent non-safety buses are automatically transferred to the 2nd preferred source of offsite power, i.e., the Standby Auxiliary Transformer.~~

~~The Standby Auxiliary Transformer also provides power to the stations Auxiliary Boiler and, if required, Cooling Tower forced cooling motors.~~

The fourth switchgear group designated Permanent Non-safety "Y" is normally connected to the ~~4,160V~~ Unit Auxiliary Transformer "Y". It also has the same ability to be connected to ~~either~~ the Unit ~~Standby~~ Auxiliary Transformer "X" or alternate Alternate AC source as previously described for the third switchgear.

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"Each of the above switchgear is normally separately tied to its own secondary winding of a dual low voltage 13,800/4,160V Unit Auxiliary Transformer through its normal incoming breaker. Should the switchgear lose its normal source, it will automatically transfer to its alternate incoming breaker to share another switchgear's Unit Auxiliary Transformer. If the the residual voltage of the 4,160V motors are in synchronism with the alternate source, a fast transfer will result. If they are out of synchronism, the transfer will be delayed until the residual voltage is 25% or less."

The Permanent Non-safety "X" and "Y" switchgears also are the alternate ~~normal~~ supply of preferred power to their respective 4,160 volt Class 1E Auxiliary Power System Safety Load Divisions I and II as described in Section 8.3.1.1.2.

These four non-Class 1E switchgear groups, with the four sources of power (preferred-1, preferred-2, main generator and AAC) and their ability to energize the Division I or II safety loads reduce the likelihood of Station Blackout.

The protective relaying for the 4,160 volt switchgear feeders and buses can be classified in four separate protection configurations. The type, size, and function of the protected equipment determines which of the schemes below will be employed.

A. Protection of large (5MVA or above) motors and (or special) transformers.

B. Protection of small motors and small transformers.

C. Protection of AC sources.

D. Protection of buses and bus feeders.

The protective schemes are designed to isolate the faulted equipment from the rest of the system, to minimize the effect of the fault, and to maximize availability of the remaining equipment. The scheme also limits the damages and the time out of service of the faulty equipment. Each scheme is designed to best achieve this for the specific equipment protected. The basic schemes consist of ground fault protection, instantaneous overcurrent and timed overcurrent protection. Other forms of protection, such as undervoltage, and reverse power flow, are provided where applicable. Each breaker in this auxiliary power system is provided with timed overcurrent protection and an anti-pump device. E

8.3.1.1.1.4 480 Volt Normal Auxiliary Power System

The 480 Volt Normal Auxiliary Power system is energized by the 4160V Normal Auxiliary Power System switchgear through 4160V to 480V transformers.

The secondary of a typical transformer is connected to a 480 volt load center bus through a 480V load center circuit breaker. Connected to the load centers are large motors, large heaters and 480 volt motor control centers located throughout the plant in areas of concentrated 480V loads.

In the application of the 480V load centers, a selective system is used whereby both the main and feeder circuit breakers have interrupting capacity greater than their required duty.

The main breakers are equipped with overcurrent trip devices having long-time and short-time delay functions, and the feeder breakers are equipped with overcurrent trip devices having long-time and instantaneous functions. Each breaker in the auxiliary power system is provided with an anti-pump device. E

8.3.1.1.2 Class 1E AC Power Systems

8.3.1.1.2.1 4,160 Volt Class 1E Auxiliary Power System

Each unit has two redundant and independent 4,160V Class 1E Auxiliary Power Systems, identified as Safety Divisions I and II. insert X → ~~which normally receive power from the 4,160V Normal Auxiliary Power System.~~ X The incoming source breakers trip upon loss of offsite ~~normal~~ power, and emergency power is provided to each of the redundant 4,160V Class 1E Auxiliary Power System Divisions by two (one per division) separate and completely independent emergency diesel generators (EDGs). I In the event of a diesel generator out of service or failure condition, the Alternate AC source can be aligned to provide emergency power to either Class 1E Safety Load Division.

Each of the redundant 4,160V safety buses is provided with undervoltage protection to monitor bus voltage. E

The under-voltage setpoint is selected such that relay operation will not be initiated during normal motor starting; however, these relays will detect loss of voltage and initiate action in a time frame consistent with the accident analysis.

All safety-related equipment in the plant requiring electrical power during a Loss of Offsite Power, Loss of Coolant Accident, or major secondary system break condition is fed from the 4,160V Class 1E Auxiliary Power System, either directly if at 4,160V or through transformers if at a lower voltage. All Engineered Safety System loads are assigned to the two 4,160V Class 1E Auxiliary Power Systems with capacities and quantities such that the failure of any component in one of the two Class 1E Auxiliary Power Systems does not affect the other system. Refer to Table 8.3.1-2 for listing of typical Class 1E equipment, loads and design ratings.

With such an arrangement of emergency diesel generators, I electrical distribution system and loads, complete redundancy of the entire Class 1E Auxiliary Power System is provided. E

INSERT X

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"The four 4,160V buses (A & C for Division I and B & D for Division II), normally receive power from their associated Safety Systems Transformer."

8.3.1.1.3.2 **Periodic Tests**

Inspection, maintenance and testing is performed in accordance with a periodic testing program. The periodic testing program is conducted so as not to interfere with unit operation. Where tests do not interfere with unit operation, system and equipment tests are scheduled with the nuclear unit in operation. The means to accomplish this testing is described below.

The 13,800V and 4,160V circuit breakers and associated equipment can be tested in service where testing does not interfere with the operation of the Unit. These circuit breakers can be "racked out" to a test position and operated without energizing the circuits. A separate feed (whose breakers are normally open) from the ~~Standby Auxiliary Transformer~~ 4160V Permanent Non-Safety Switchgear to each Class 1E Safety Load Division is provided to facilitate maintenance and testing of the normal source breakers feeding each Division.

The 480V circuit breakers, motor contactors and associated equipment can be tested in service by opening and closing the circuit breakers or contactors. Transfers to the various emergency power sources can be tested on a routine basis to prove the operational ability of these systems. E

In compliance with General Design Criterion 18 and the intent of Regulatory Guide 1.22, the Class 1E Auxiliary Power System design is such that inspection, maintenance and periodic testing can be carried out with a minimum of interference with operation of the nuclear unit. Unit design includes two completely redundant 4,160V, four redundant 480V, and four redundant 120 Volt Class 1E Auxiliary Power busses. Testing during reactor operation can be accomplished by allowing one system to be taken out of service for testing. Breakers can be racked out to the test position while the system is undergoing test. Continuous indication of unavailable systems is provided in the control room.

The generator power circuit breaker (PCB) periodic test program includes ~~load~~ open/close measurements, and dielectric tests.

Testing of protective relays is performed on a periodic basis. Testing facilities are provided to meet the capability for testing in compliance with General Design Criterion 18 and the intent of Regulatory Guide 1.22. Relay sensors such as current transformers are tested before initial installation and unit operation and periodically thereafter. These protective devices are in service during normal operation. The preoperational tests for the protective relaying system verify the continuity of the system and the condition of all the components. The methods used to accomplish this are as follows:

A. All relays and other momentary duty type operating devices associated with the protective relaying of the onsite power system are tested to determine individual performance characteristics, and assure repeatability of design settings, under various simulated conditions. This ensures device integrity.

B. All relay sensors such as current and potential transformers are tested for correct and reliable outputs.

C. All interconnecting wiring and cabling is inspected for proper installation and connections.

D. All protective relaying systems are tested under necessary simulated conditions to verify correct operation in preferred, alternate and abnormal modes. E

The 120V AC Vital Power System is normally powered from inverters which are in use during normal operation. The continuous operation of the inverters is indicative of their operability and functional performance since accident conditions will not substantially change their load.

8.3.1.1.4 Class 1E Emergency Diesel Generators I

Each Division of the 4,160V AC Class 1E Auxiliary Power System is supplied with emergency standby power from an independent emergency diesel generator. The emergency diesel generator is designed and sized with sufficient capacity to operate all the needed engineered safety feature and emergency shutdown loads powered from its respective Class 1E Safety Division bus. E I E

Each emergency diesel generator is designed to attain rated voltage and frequency within 20 seconds and to begin accepting sequenced loads after receipt of a start signal to meet the response times assumed in Chapter 15 analyses. Refer to Table 8.3.1-2 for loading sequence and bases. The characteristics of the generator exciter and voltage regulator provide satisfactory starting and acceleration of sequenced loads and ensures rapid voltage recovery when starting large motors. The generator voltage and frequency excursions between sequencing steps are in compliance with the intent of Regulatory Guide 1.9. I E

Each emergency diesel generator and its associated auxiliaries are installed in separate rooms and are protected against tornadoes, external missiles, and seismic phenomena. The diesel rooms are protected with firewalls which are designed to prevent the spread of fire. ~~from one diesel room to the redundant diesel room.~~ Refer to Section 9.5.9 for a description of the diesel room sump pump. I

The AAC is not normally nor automatically directly connected to any Class 1E Safety Load Division. However, it can be manually aligned to power one Safety Load Division via one Permanent Non-Safety Bus, to accommodate a emergency diesel generator failure or out-of-service condition. E I

8.3.1.1.5.1 AAC Starting and Loading E

The AAC is designed to start automatically within ten minutes from the onset of a LOOP event. It is then available for loading if either of the 4,160V Permanent Non-Safety Load Busses X and Y become de-energized. Automatic connection and sequential loading of the X and/or Y permanent non-safety loads will occur utilizing a sequencer design similar to that described in 7.3.1.1.2.3. I E

8.3.1.1.5.2 AAC Instrumentation and Controls

The instrumentation and controls necessary to start and run the AAC are powered from a dedicated local 125V DC battery.

Various monitoring and control devices are provided locally and in the control room to give the operator control and operational status information. The following typical parameters are monitored and/or alarmed:

A. Lube oil temperatures and pressures

B. Bearing temperatures

C. Cooling temperatures and pressures

D. Generator parameters and status

E. Speed

F. Starting air pressure

G. Control mode status (standby, starting, running, local).

8.3.1.1.5.3 AAC Auxiliary Support Systems

A. Fuel System and Supply

The AAC is equipped with redundant fuel systems. Sufficient fuel is stored on site to support 24 hour operation at rated load.

comprised of diverse relaying. Tripping of the independent lockout relays is achieved through a coincidence of like trip signals.

This requirement prevents a false trip of the lockout relays due to a malfunction of one relay. The scheme also allows for testing and maintenance of each channel without causing a false trip and without removing the protection from the system. The inherent quality of this scheme is that each primary channel provides the redundancy needed for proper operation in case one relay fails and assurance of not tripping due to false operation of one relay.

~~Non-safety buses feeding loads required for unit operation only are provided with an undervoltage protection scheme design to protect the loads against damage due to sustained operation under degraded voltage conditions and shed all major loads under loss-of-voltage conditions.~~

Class 1E Division buses and permanent non-safety buses ~~feeding permanent non safety loads~~ are provided with separate bus voltage monitoring and protection schemes for degraded voltage and loss of voltage conditions, respectively. These schemes are designed according to the recommendations of IEEE Standard 741 "IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Systems." Two separate time delays are selected for degraded voltage protection as recommended in IEEE Standard 741, Appendix A. Based on the automatic bus transfer sequences adopted, a time delay is provided for loss of voltage relay actuation to preclude unnecessary starting of the onsite standby power sources during the transfer sequences. The undervoltage protection schemes use coincidental logic (e.g., two out of three phases) to avoid spurious trips of the offsite power sources.

The relay zones in the Onsite Protection System overlap to maintain protection throughout the system. Any fault condition in a particular tripping zone trips the circuit breakers in that zone by its associated protective relays.

8.3.1.1.7 Monitoring Instrumentation and Controls for Onsite Power System

The monitoring instrumentation associated with the Onsite Power System provides a reliable source of information in the control room and protective functions for major components. The instrumentation provides quantitative values and status conditions for the operator in the control room. This instrumentation provides the operator with the information necessary for efficient operation of the unit. The

8.3.1.1.8 **Design Bases for Class 1E Motors**

As a minimum, Class 1E motors are capable of accelerating their loads within the required time with a starting voltage as low as 75% of rated motor voltage. E

When operated under nominal conditions, the plant motors have a continuous power rating greater than the maximum power rating of the driven equipment. Service factor requirements are in accordance with NEMA Standard MG 1 "Motors and Generators" - Section MG 1-12.47. I

Except where specified otherwise, medium voltage motors which are required to operate continuously during normal plant operation, are designed for Class B temperature rises and provided with Class F insulation systems.

Medium voltage motors which are required to operate continuously during normal plant operation are provided with thermocouples or resistance temperature devices to measure winding and bearing temperatures. E

8.3.1.2 <u>Analysis</u>

The 4,160V AC and 480V AC Safety Auxiliary Power Systems (Divisions I and II) are Class 1E systems, and as such are designed to meet the requirements of General Design Criteria 17 and 18, and the intent of NRC Regulatory Guides 1.6, 1.9, 1.32, 1.63, 1.81, and 1.106 as discussed below. A failure modes and effects analysis for the onsite power system is presented in Table 8.3.1-1.

8.3.1.2.1 **Compliance with General Design Criterion 17 and Regulatory Guide 1.32**

THREE ~~Two~~ separate circuits from the transmission network are normally available to the Class 1E Auxiliary Power Systems.

The separation of the three ~~two~~ independent circuits at the offsite voltage level is maintained by the switchyard power circuit breakers. Each circuit is separately connected through transformers and breakers to the redundant 4160V Safety Switchgear or the 4,160V Permanent Non-safety switchgear, ~~which in turn are~~ connected through double isolation feeder breakers to the redundant Division I or II 4160V Class 1E Safety Division switchgear. ~~Since each of the supplies is normally available within seconds following the tripping of the reactor and the opening of the generator breakers, the~~ The requirements of GDC 17 and the intent of guidance in Regulatory Guide 1.32 are fully met.

The 4160V Permanent Non-Safety Switchgears, can alternately be

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(Permanent Non-Safety Switchgear associated with that division

In the event that one of the two 50% capacity Safety Systems ~~Unit Auxiliary~~ transformers is out of service, the 4,160V Class 1E Safety Division System switchgear supplied from that transformer will be supplied from the ~~Standby Auxiliary transformer~~, thereby maintaining two independent circuits to the Class 1E Divisions I and II during this period. In the event that the unit main transformers are out of service, both of the 4,160V Class 1E Divisions ~~can be supplied from the remaining single independent Standby Auxiliary Transformer circuit~~ will be unaffected and continue to be fed from the Safety Systems Transformers.
The Onsite Power System is designed to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of the unit generator, the transmission network, or the onsite electric power supplies.

8.3.1.2.2 **Compliance with General Design Criterion 18**

Provisions are made for periodic testing of all important components of the Class 1E AC power systems. Further provision is made for periodic testing of the emergency diesel generators to assure their capability to start and to accept loads within design limits. Electric power systems important to safety are designed to allow periodic testing to the extent practical. Included in the system design is the capability to periodically test the operability and functional performance of these systems as a whole and under conditions as close to design as practical. Staggered tests may be employed to avoid the testing of redundant equipment at the same time. E

The 4,160V circuit breakers and associated equipment are tested in-service by opening and closing the breakers so as not to interfere with the operation of the unit. The 480V breakers, motor starters, and associated equipment are also tested in-service by opening and closing the breakers and contactors so as not to interfere with unit operation. Additionally, the protective relaying associated with the 4,160V and 480V Safety Auxiliary Power System Divisions are inspected, tested, and maintained on a routine basis.

8.3.1.2.3 **Compliance with Regulatory Guide 1.6**

The design of the Class 1E AC power systems complies with the intent of independence requirement of Regulatory Guide 1.6.

The electrically powered Class 1E AC loads are separated into two redundant and completely independent divisions for the unit. There are no direct automatic or manual ties between redundant divisions.

TABLE 8.3.1-1

(Sheet 1 of 7)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ONSITE POWER SYSTEMS

Component	Malfunction	Resulting Consequences
1. Isolated phase bus from unit main transformer to the generator breaker or to the unit auxiliary transformer. or Unit Auxiliary transformer or Unit Main Transformer or Unit Auxiliary Transformer Secondary Non-Seg Bus	Loss of one due to a fault	(a) The faulted equipment is isolated by protective relaying and protective equipment. (b) The other ~~independent preferred offsite~~ circuit from Preferred Switchyard Interface I remains unaffected. ~~(c) Automatic reactor trip occurs.~~ ~~(d) The unit generator automatically trips and its breaker opens.~~ d ~~(e)~~ ~~The~~ The 13.8kV Non-Safety, 4.16kV Non-Safety and the 4.16kV Permanent Non-Safety Auxiliary System switchgear supplied from the faulted circuit is connected in a automatic rapid bus transfer to the Standby Auxiliary transformer in the second independent circuit and Class 1E auxiliaries continue to receive uninterrupted offsite power. (from the Safety Systems Transformers ~~(f) Emergency diesel generator and AAC source automatically start.~~

E

(c) The affected generator breaker opens and the unit runs back so as not to overload the remaining transformers and isolated phase bus.

TABLE 8.3.1-1 (Cont'd)

(Sheet 2 of 7)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ONSITE POWER SYSTEMS

Component	Malfunction	Resulting Consequences
2. ~~Standby~~ Safety ~~Auxiliary~~ Systems Transformer	Loss due to a fault	(a) The faulted equipment is isolated by protective relaying and protective equipment. (b) The ~~other independent preferred~~ offsite circuit from Preferred Switchyard Interface II remains unaffected. The other Safety Systems Transformer is unaffected. (c) No effect on unit power generation ~~or Essential Safety buses, since not normally connected to onsite system~~. ~~(d) Loss of power source to Auxiliary Boiler (and, if supplied, Cooling Tower Fans).~~ E (d) The associated diesel generator starts and loads on the affected switchgears are sequenced.
3. Isolated phase bus connecting the generator circuit breaker and the unit generator or Unit generator	Loss due to a fault	(a) Generator breakers trip. (b) The unit turbine-generator is tripped automatically. (c) Reactor Power Cutback System initiation. (d) All unit and Class 1E auxiliaries continue to receive uninterrupted offsite power from the Unit Auxiliary transformers and Safety Systems Transformers.
INSERT B→ ~~4. Generator circuit breaker~~	~~Loss of one pole of the breaker.~~	~~(a) The other two poles of the breaker trip.~~ ~~(b) The Non-Safety Auxiliary System switchgear are supplied from the Unit Main Transformers, all unit and Class 1E auxiliaries continue to receive uninterrupted power through the preferred offsite circuit~~s.

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INSERT B

"4.	Generator Circuit Breaker	Breaker Fault, Failure, or Pole Disagreement	(a) Faulted or failed equipment is isolated by protective relaying and protective equipment. (b) The other circuit from the preferred switchyard interface I remains unaffected. (c) The unit turbine generator is tripped automatically. (d) Reactor power cutback system initiation. (e) The 13.8KV Non-Safety, the 4.16KV Non-Safety and the 4.16KV Permanent Non Safety Auxiliary System switchgears supplied from the faulted circuit is connected in a automatic rapid bus transfer to the alternate unit auxiliary transformer. Class 1E auxiliaries continue to receive uninterruptedoffsite power from the safety systems transformers."

TABLE 8.3.1-1 (Cont'd)

(Sheet 3 of 7)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ONSITE POWER SYSTEMS

Component	Malfunction	Resulting Consequences	
~~4. (Cont'd)~~		~~(c) The unit generator is tripped automatically.~~	
		~~(d) Reactor Power Cutback System initiation.~~	
5. Isolated Phase Bus Cooling System	Loss of bus cooling	(a) No immediate consequence. The unit ~~and Class 1E~~ auxiliaries continue to receive an uninterrupted flow of power from the Unit Auxiliary Transformers. However, continued unit operation is dependent upon bus design capacities ~~with and~~ without forced cooling.	E
6. Unit Auxiliary Transformers Cooling System	Loss of one of the cooler banks	(a) No immediate consequence. The unit ~~and the Class 1E~~ auxiliaries continue to receive an uninterrupted flow of power from this source. However, continued transformer ~~and unit~~ operation is dependent upon its rated design capacities ~~with and~~ without cooling.	
7. Unit Main Transformer Cooling System	Loss of one of the cooler banks	(a) No immediate consequence with step-up transformer at full load. The operator must reduce load to within ~~70% of~~ the transformer ~~forced~~ manufacturer's recommendations ~~cooling rating and maintain the self-cooled rating of the transformer.~~ The unit and the Class 1E auxiliaries continue to receive an uninterrupted flow of power.	

TABLE 8.3.1-1 (Cont'd)

(Sheet 4 of 7)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ONSITE POWER SYSTEMS

Component	Malfunction	Resulting Consequences
~~8. 13,800V Non-Safety Auxiliary System switchgear source breaker~~	~~Breaker fault~~	~~(a) The unit trips automatically due to Reactor Coolant Pump coastdown.~~ ~~(b) No affect on the 4,160V onsite power system.~~
~~9.~~ 8. ~~4,160V Non-Safety Auxiliary Power System Switchgear breaker~~	~~Breaker fault~~	~~(a) Loss of normal source to 4,160V Non-Safety Bus. Requires unit power reduction to the capacity supported by remaining non-safety auxiliaries. May cause Reactor Power Cutback or unit to trip.~~ ~~(b) No effect on Permanent Non-Safety or Class 1E Safety Division loads.~~
~~INSERT C~~ INSERT D ->		
9 ~~10.~~ 13,800V Non-Safety Auxiliary System switchgear bus or switchgear	Bus shorted or Breaker fault	(a) The switchgear source breaker trips. (b) The plant will experience a reactor trip due to the loss of reactor coolant pumps. (c) No effect on 4160V NON SAFETY, 4160V Permanent Non-Safety or Class 1E Safety Division loads.
~~11. 4,160V Non-Safety Source Feeder cable to the 4,160V Permanent Non-Safety switchgear bus or switchgear~~	~~Fault~~	~~(a) The appropriate 4,160V Non-Safety Auxiliary System and 4,160V Class 1E Safety Division switchgear breakers trip. Sufficient redundant auxiliaries remain operable from the redundant Class 1E switchgear Division for the safe shutdown of the reactor.~~

E

INSERT E, F, G, H, I, J, K, L, M, N, O, P, Q, R, S, T, U, V

RAI 430.9

INSERT D

"8.	13,800V Non-Safety Auxiliary System Switchgear Source Breaker	Breaker Fault or Failure	(a) The unit trips automatically due to Reactor Coolant Pump coastdown. (b) Protective trips isolate the appropriate Main/ Unit Auxiliary Transformer zone. (c) Associated 13.8KV and 4.16KV Non-Safety and 4.16KV Permanent Non-Safety switchgear (other than the one with the breaker fault or failure) fast transfer to their alternate source. (d) No effect on Class 1E safety division loads."

RAI 430.9

INSERT E

"10.	4160V Non Safety Auxiliary Power System Switchgear Source Breaker	Breaker Fault or Failure	(a) Protective trips isolate the appropriate Main/ Unit Auxiliary Transformer zone.
			(b) Associated 13.8KV and 4.16KV Non-Safety and 4.16KV Permanent Non-Safety switchgear (other than the one with the breaker fault or failure) fast transfer to their alternate source.
			(c) The unit runs back so not to overload the unit main transformer and isolated phase bus.
			(d) Loss of normal source to 4160V Non-Safety bus may require further power reduction to the capacity supported by remaining non-safety auxiliaries. May cause Reactor Power Cutback or unit to trip.
			(e) No effect on Class 1E safety division loads."

RAI 430.9

INSERT F

"11. 4160V Non-Safety System Switchgear	Bus shorted or feeder breaker fault	(a) The switchgear source breaker trips. (b) Loss of normal source to 4,160V Non-Safety Bus. Required unit power reduction to the capacity supported by remaining non-safety auxiliaries. May cause Reactor Power Cutback or unit to trip. (c) No effect on Permanent Non-Safety or Class 1E safety division loads."

RAI 430.9

INSERT G

"12. 4160V Non-Safety System Switchgear feeder 4160V load cables	Fault on one	(a) The associated 4160V feeder breaker trips and isolates the fault from the system. Remaining loads should not be affected. (b) Loss of source to 4160V Non-Safety load may require further power reduction to the capacity supported by remaining non-safety auxiliaries. May cause Reactor Power Cutback or unit to trip."

RAI 430.9

INSERT H

"13. 4160/480 Volt Non-Safety Load Center Transformer or its Feeder Cables or 480V Non-Safety load center source breaker	Fault on one	(a) The associated 4160V feeder breaker trips and isolates the fault from the system. (b) The load center is deenergized. (c) The 480V Non-Safety motor control centers dead bus transfer to their alternate source."

RAI 430.9

INSERT I

"14. 480 Volt Non-Safety load Center bus or 480 Volt Non-Safety Load Center Feeder Breaker	Fault	(a) The load center source circuit breaker trips. (b) The associated 480V loads are deenergized (c) The associated 480V Non-Safety motor control centers dead bus transfer to their alternate source."

RAI 430.9

INSERT J

"15. 480 Volt Non-Safety load Center Feeder Cable or 480 Volt Non-Safety Motor Control Center Source Breaker	Fault	(a) The load center feeder breaker trips. (b) The load or motor control center remains deenergized."

RAI 430.9

INSERT K

"16.	480 Volt Non-Safety Motor Control Center Bus or 480 Volt Non-Safety Motor Control Center Source Feeder Breaker	Fault	(a) The motor control center source breaker trips."

RAI 430.9

INSERT L

"17.	480 Volt Non-Safety Motor Control Center Feeder Cable	Fault	(a) The motor control center feeder breaker trips."

RAI 430.9

INSERT M

"18.	4160V Permanent Non-Safety Auxiliary Power System Switchgear Source Breaker	Breaker Fault or Failure	(a) Protective trips isolate the appropriate Main/ Unit Auxiliary Transformer zone. (b) Associated 13.8KV and 4.16KV Non-Safety and 4.16KV Permanent Non-Safety switchgear (other than the one with the breaker fault or failure) fast transfer to their alternate source. (c) The unit runs back so not to overload the unit main transformer and isolated phase bus."

RAI 430.9

INSERT N

"19.	4160V Permanent Non-Safety System Switchgear	Bus shorted or feeder breaker fault	(a) The switchgear source breaker trips. (b) Sufficient redundant auxiliaries remain operable from the redundant Permanent Non-Safety System switchgear. (c) No effect on Class 1E safety division loads."

RAI 430.9

INSERT O

"20.	4160V Permanent Non-Safety System Switchgear 4160V load feeder cables	Fault on one	(a) The associated 4160V feeder breaker trips and isolates the fault from the system. Remaining loads should not be affected. Sufficient redundant auxiliaries remain operable from the redundant Permanent Non-Safety System."

TABLE 8.2-1 (Cont'd)

(Sheet 2 of 2)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE OFFSITE POWER SYSTEM

Component	Malfunction	Resulting Consequences	
3. (Continued)			E
or			
Circuit from switchyard to main unit transformer Interface I		(c) If on-line, the unit main turbine generator is automatically run back to the appropriate rated output and all unit ~~and Class 1E~~ auxiliaries continue to receive uninterrupted power.	I, E
or			
Main Unit Transformer		(d) If the unit main turbine generator is off line, one of the two independent offsite circuits is available for the Permanent Non-safety ~~and Class 1E Division~~ auxiliaries via ~~the Standby~~ an alternate Unit Auxiliary Transformer.	I, E

INSERT E, AA →

NOTE: The Offsite Power System shall be protected such that it is unaffected by failures in the Onsite Power System.

RAI 430.9

INSERT P

"21.	4160/480 Volt Permanent Non-Safety Load Center Transformer or its Feeder Cables or 480V Permanent Non-Safety load center source breaker	Fault on one	(a) The associated 4160V feeder breaker trips and isolates the fault from the system. (b) The load center is deenergized. (c) The 480V Permanent Non-Safety motor control centers dead bus transfer to their alternate source. (d) Should any load center loads be lost, sufficient redundant auxiliaries remain operable from the redundant Permanent Non-Safety System."

RAI 430.9

INSERT Q

"22.	480 Volt Permanent Non-Safety Load Center bus or 480 Volt Permanent Non-Safety Load Center Feeder Breaker	Fault	(a) The load center source circuit breaker trips. (b) The associated 480V loads are deenergized. (c) The associated 480V Permanent Non-Safety Motor Control Centers automatically transfer to their alternate source."

RAI 430.9

INSERT R

"23. 480 Volt Permanent Non-Safety load Center feeder Cable or 480 Volt Permanent Non-Safety Motor Control Center Source Breaker	Fault	(a) The load center feeder breaker trips. (b) The load or motor control center remains deenergized."

RAI 430.9

INSERT S

"24. 480 Volt Permanent Non-Safety Motor Control Center Bus or 480 Volt Permanent Non-Safety Motor Control Center Feeder Breaker	Fault	(a) The motor control center source breaker trips."

RAI 430.9

INSERT T

"25.	480 Volt Permanent Non-Safety Motor Control Center Feeder Cable	Fault	(a) The motor control center feeder breaker trips. Sufficient redundant auxiliaries remain operable from the redundant Permanent Non-Safety System."

RAI 430.9

INSERT U

"26. 4160V AAC Source feeder cable or breaker	Fault	(a) If the AAC Source is connected, the 4160V Permanent Non-Safety switchgears are deenergized. (b) At the operators discretion, critical Permanent Non-Safety loads may be backfed from the diesel generator provided sufficient load shedding has taken place."

RAI 430.9

INSERT V

"27.	4160V Class 1E Safety Auxiliary Power System Division Switchgear Source Breaker	Breaker Fault or Failure	(a) Safety Systems Transformer zone trips isolating the fault.
			(b) Affected 4160V Class 1E switchgear is deenergized. Associated 480V buses are also deenergized.
			(c) The associated diesel generator starts and loads on unfaulted 4160V class 1E switchgear are sequenced on.
			(d) Sufficient redundant auxiliaries remain operable from the redundant class 1E Safety Power System Division for safe shutdown of the reactor."

TABLE 8.3.1-1 (Cont'd)

(Sheet 5 of 7)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ONSITE POWER SYSTEMS

Component	Malfunction	Resulting Consequences
		(b) Loss of normal source to 4,160V Permanent Non-Safety bus causes it to fast transfer to Standby Auxiliary transformer. Onsite EDG automatically starts.
12. 4,160V Permanent Non-Safety Auxiliary Power System switchgear source breaker or cable from Standby Auxiliary Transformer	Fault	(a) Normally no effect. (b) If the Standby Auxiliary transformer is the Source connected and the Unit Auxiliary Transformer source is not available, the appropriate source breakers open to isolate the affected Class 1E Safety Division Bus. The AAC and EDG start and are load sequenced.
13. 4,160V AAC Source or feeder cable or breaker to the 4,160V Permanent non-safety switchgear	Fault	(a) If the AAC Source is connected, consequences similar to 11.(a), (b) above.
~~14.~~ 23. 4,160V Class 1E Safety Auxiliary Power System Division switchgear or Feeder Breakers or 4,160V Safety Division Source feeder cables or breakers from 4,160V Permanent non-Safety switchgear	Fault	(a) Source breakers trip and the affected 4,160V Class 1E buses ~~Division switchgear is~~ deenergized. Loss of redundant 1E 480V ~~loads~~ (Channels A, C or B, D) associated with division. Sufficient redundant auxiliaries remain operable from the redundant Class 1E Safety Auxiliary Power System Division for the safe shutdown of the reactor.

E

supplying power under blackout conditions

TABLE 8.3.1-1 (Cont'd)

(Sheet 6 of 7)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ONSITE POWER SYSTEMS

Component	Malfunction	Resulting Consequences
~~25~~ ~~30~~ 29 ~~15.~~ 4,160V Safety Division Emergency Diesel Generator ~~or its breaker~~	Fault	(a) If the EDG source is ~~connected~~ supplying power under blackout conditions, the affected 4,160V Safety Division is deenergized until the fault is cleared and the AAC source can be manually aligned to re-energize the division. (b) Sufficient redundant auxiliaries remain operable from the redundant Class 1E Safety Power System Division.
30 ~~16.~~ 4,160V Class 1E Safety Auxiliary Power System Division switch-gear feeder cables or 4,160/480 Volt 1E load center transformer or 480 V Class 1E load center source breaker	Fault on one	(a) The associated load feeder breaker trips and isolates the fault from the system. Remaining Division loads should not be affected. ~~However, if the fault trips the Division's source breaker,~~ Sufficient redundant auxiliaries remain operable from the redundant Class 1E Safety Power System Division ~~and channels~~ for safe shutdown of the reactor. E
31 ~~31~~ ~~17.~~ 480 Volt Class 1E load center bus or 480 Volt Class 1E load center feeder breaker	Fault	(a) The load center source circuit breaker trips. Sufficient redundant auxiliaries remain operable from the redundant Class 1E Safety Power System Division for the safe shutdown of the reactor.

TABLE 8.3.1-1 (Cont'd)

(Sheet 7 of 7)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE ONSITE POWER SYSTEMS

Component	Malfunction	Resulting Consequences
32 ~~18~~. 480 Volt Class 1E load center feeder cable or 480 Volt Class 1E motor control center bus	Fault	(a) The load center feeder breaker trips. Sufficient redundant auxiliaries remain operable from the redundant Class 1E Safety Power System Division for the safe operation of the reactor.
33 ~~19~~. 480 Volt Class 1E motor control center feeder cable	Fault	(a) The motor control center feeder breaker trips. Sufficient redundant auxiliaries remain operable from the redundant Class 1E Safety Power System Division for the safe operation of the reactor. E

Question 430.15

In Section 8.3.4.1C(6) it is stated that if offsite power is the source of power to an emergency bus when ESFAS is generated, the EDG loads which are appropriate to the particular ESFAS shall be started either immediately or by sequencing on the offsite powered emergency bus. In regard to the option to sequence safety loads when preferred power is available, the staff believes that the load sequencer represents an additional source of unreliability for the "preferred" power source. Additionally, since the sequencer is common to the offsite power source and the onsite power source (diesel generator), a failure of this unit could potentially result in total loss of ac power to that bus. Therefore, the staff requires that either two load sequencers, one for offsite power and one for the onsite power source be provided, or this option should be deleted from the SSAR.

Response 430.15

The referenced Section 8.3.4.1C(6) no longer exists, however, sequencing of safety loads in response to an ESFAS is described in Section 8.3.1.1.4.6 "Load Shedding and Sequencing" and in Section 7.3.1.1.2.3 "CCS Diesel Load Sequencing".

Two load sequencers are provided in the design by incorporating redundant sequencers in each division rather then separate sequencers for offsite and onsite power.

Each Division of the safety equipment is provided with two sequencers configured in a redundant pair. This protects the flow path independence of each Division against a single failure.

Within each Division two load sequencers are provided by using two CPU's in a hot standby configuration. If one of the CPU's fails, automatic transfer to the hot standby CPU is initiated thereby protecting against a single failure.

Further defense in depth is provided by the alternate AC power source (gas turbine generator) which can be aligned to feed power to either the Division I or Division II safety bus in the event of failure of either of the EDG's or the preferred power source.

Question 430.25

It is stated in Section 7.3.1.1.2.3.A that EDG loading sequencer initiation logic monitors various plant electrical buses to determine when abnormal power conditions exist. When the right combinations of bus abnormalities are present, a loss of offsite power signal is generated to auto start EDG and load shed the necessary plant equipment. Please identify the buses which are monitored by the load sequencer logic to initiate EDG start and load shed signal.

Response 430.25

The Diesel Load Sequencer continuously monitors the Class 1E 4.16 KV Division I and Division II buses for an undervoltage condition using four undervoltage relays, two on each bus. Undervoltage on these buses starts the EDG and initiates load shedding.

Question 430.26

Recent incidents at operating plants (IN 91-06) have revealed that the design of the load sequencing circuits that provide an automatic start signal upon sensing bus undervoltage and the interface between these circuits and the interlocks in the EDG air start system are such that the circuits will lock-up whenever an EDG trip results in an undervoltage on the associated safety bus. This would necessitate proper operator action to reset the locked-up circuits in order to restart the EDG promptly. These incidents have raised concerns regarding the understanding of EDG and load sequencer control circuits and their interfaces, and the adequacy of procedures for restarting EDGs following unexpected trips. Successful mitigation of the effects of accidents or transients and maintenance of the overall reliability of EDGs depend upon operations personnel having (1) sufficient knowledge of the associated instrumentation and controls; and (2) the ability to recognize and reset a lock-up condition.

Provide a discussion of your EDG load sequencer design and identify the conditions that will cause the load sequencer controls to lock-up and prevent subsequent restart of the EDGs. In addition, provide a detailed plan of the level of training proposed for your operators to assure optimum availability of the EDGs.

Response 430.26

The diesel load sequencer functional design is described in Section 7.3.1.1.2.3 of CESSAR-DC. The current DLS functional design has no conditions that will cause the load sequencer controls to lock-up and prevent subsequent restart of the EDGs, however, the detailed logic design including interlocks and permissive conditions for the EDG start and/or restart after trip functions is part of FOAKE since it requires detailed manufacturer's technical data on the selected diesel generator.

To ensure that lock-up conditions have not been introduced during implementation of the DLS functional design, the independent design verification and validation program will include factory and preoperational testing on an integrated system level to assure that no credible scenarios, including those described in IN 91-06, are present in the implemented design.

Training of operations and maintenance personnel is an Owner/Operator responsibility.

Question 430.49(8.3.2)

Section 8.3.2.1.2.1 states that Division I and II include an additional battery, battery charger, DC distribution center, inverter and AC panel board for their respective divisions. Describe the use of the additional equipment provided for each safety-related division. In addition, specify the criteria to be used for locating battery chargers and main distribution panels associated with each battery set.

Response 430.49(8.3.2)

The additional battery, battery charger, DC distribution center, inverter and AC panel board are provided for each division to satisfy EPRI ALWR Utility Requirements Document requirement 7.3.1.4, which requires that provisions be made to permit connecting each bus to a standby, backup DC source. The additional equipment is cross-connected to both safety channels of a division (A and C for Division I; B and D for Division II) through normally opened breakers. This allows either channel or the division inverter to be powered from the backup battery, battery charger, and DC distribution center.

The criteria for locating the battery chargers and main distribution panels, per section 8.3.2.2.1 of CESSAR-DC, complies with the intent of IEEE Standard 308-1980 as augmented by Regulatory Guide 1.32, and, per section 8.3.1.2.5, the Onsite 1E AC Electrical Power Systems redundant division equipment is physically and electrically independent from each other in accordance with IEEE Standard 308-1980.

Question 430.50(8.3.2)

Figure 8.3.2-2 of SSAR does not show a backup AC source to allow normal system operation in case of a failure or unavailability of a single inverter. Please modify this figure to include a backup source for each Class 1E vital bus and provide the source of this backup supply. It is our position that the backup source to the vital instrument buses be qualified 1E and be connected to a Class 1E power source. This will ensure that the Class 1E vital loads are not jeopardized by connection to an unqualified power source during periods when they are fed from the backup source.

Response 430.50(8.3.2)

Figure 8.3.2-2 is a functional overview drawing. This figure is not intended to show actual equipment construction or layout. The functional block labeled "CH. X INVERTER with STATIC SW/MANUAL BYPASS SW" will actually be composed of several individual components. The inverter is separate from the transfer switch. Thus, the backup AC power is provided via a transfer switch downstream of the inverter circuitry and, thus able to successfully supply the Vital AC Instrument buses in the event of inverter failure or unavailability. The back-up AC power is Class 1E supplied from the 4.16 Safety Load Buses.

Question 440.23

The currently available CE EPGs (CEN-152) may not be applicable to the System 80+ design. Provide a discussion for the necessary modifications made to the existing EPGs applicable to System 80+.

Response 440.23

Since the System 80+ design, as an evolutionary ALWR, is functionally similar to the System 80 design, the guidelines contained in CEN-152, Combustion Engineering Emergency Procedure Guidelines (EPGs), remain valid for use as a generic basis for plant specific procedure generation at utilities with a System 80+ design. The EPGs were developed on a generic basis by the NSSS supplier since it was recognized that the details of the Nuclear Steam Supply Systems and Balance of Plant designs varied from one plant to another. The guideline structure was designed to accommodate revisions necessary for plant specific design features to ensure operational compatibility. CEN-152 currently contains a methodology for incorporating design features which vary from plant to plant. This methodology is described in Section 13.0, "Implementation Guidance," and will be used on System 80+ plants.

Several functional enhancements were made in the System 80+ design. The following is a list of System 80+ enhancements:

Enhancements to the Safety Injection System,

Additional emergency feedwater pumps,

Cross connection of emergency feedwater trains,

Interchangability of containment spray and shutdown cooling pumps,

In-containment Refueling Water Storage Tank,

Safety Depressurization System,

Cavity Flooding System,

Alternate AC Power Supply

Some of the functional enhancements made in the System 80+ design would require modifications to CEN-152. Two examples, and the recommended approaches for incorporation into CEN-152, are cited below.

Safety Depressurization System

As stated in CESSAR-DC Section 6.7, the rapid depressurization function is part of the Safety Depressurization and Vent System (SDVS). Rapid depressurization of the Reactor Coolant

440.23

System, in conjunction with the Safety Injection System (SIS), will be used as a last resort for the total loss of feedwater beyond design basis event (see response to RAI 722.15).

In the CEN-152 "Loss of Feedwater" recovery guideline, the following changes will be made to accommodate design enhancements made to System 80+:

> Motor-operated valves will replace the generic power operated valves, references to low pressure safety injection pumps will be deleted, direct vessel safety injection will be employed instead of cold leg injection,and cooling to the in-containment refueling water storage tank will be required within a defined timeframe during rapid depressurization. Further, a maximum time in which to operate the rapid depressurization valves will be specified, to ensure that the core remains covered.

Enhancements to the Safety Injection System

As stated in CESSAR-DC Section 6.3, the System 80+ SIS consists of four mechanical trains, direct injection to the vessel, and a suction line from the in-containment refueling water storage tank. Furthermore, the System 80+ SIS does not employ low pressure safety injection pumps. Switchover from an outside refueling water tank to the containment sump is no longer necessary during a rapid RCS depressurization. The in-containment refueling water storage tank is the source of water for the safety injection pumps for all post-accident core cooling modes. As a result of the SIS enhancements made for the System 80+ design, the recovery guidelines contained in CEN-152 will remain functionally similar, with specific modifications as noted above.

Some of the functional enhancements made to the System 80+ design will not require modifications of CEN-152. An example of this type of a design enhancement is shown below.

Alternate AC Power Supply

As stated in CESSAR-DC Section 8.3.1.1.5, the System 80+ electrical distribution system employs a gas turbine as a diverse, alternate AC power source for use in the unlikely event of a station blackout. Further, this alternate AC source can be used to power one 4160 volt vital bus. CEN-152 already accounts for an alternate AC source (see CEN-152, Figure 11-9a and the Functional Recovery Guideline "Maintenance of Vital Auxiliaries"); therefore, no modification to the guidelines is required.

440.23

In summary, the currently available Emergency Procedure Guidelines (CEN-152) are applicable to the System 80+ design. Certain modifications will be made to the plant procedures to account for System 80+ design enhancements. These modifications will be made within the current CEN-152 structure to ensure operational compatibility with the System 80+ design, and will have an appropriate analytical basis. These modifications are not anticipated to invalidate the current basis or intended purpose of CEN-152, which is to provide guideline level operator guidance for use in the generation of plant specific operating procedures. This conforms with their use as specified in current industry and regulatory guidance.

Question 440.105

Discuss the radiological consequences for a small break of the letdown line outside containment assuming that the event does not result in actuation of safety-grade alarms to alert the operators.

Response 440.105

The radiological consequences of a small break will be bounded by the consequences of a large break of the letdown line, which are discussed and evaluated in Section 15.6.2 of the CESSAR-DC. System 80+™ incorporates design features which would limit the radiological consequences of a small break discussed below.

The System 80+™ design provides several means of detection of leakage outside containment. These include floor drain sump level indication in the Nuclear Annex and radiation monitors. Non-safety related radiation monitors in the Nuclear Annex could be utilized by the operators to provide indication of a small break failure of the letdown line. For instance, local area radiation monitors in the Nuclear Annex would provide indication of a spill in the general area and process monitors would provide indication of a release to the environment. Based on indications from these monitors, the operators could take appropriate action to isolate the letdown line to minimize the radiological consequences to the public and plant personnel.

Also, radioactive liquids released into the Nuclear Annex would be contained within the building. Floor drains in the Nuclear Annex would collect the radioactive liquid and route it to the Liquid Waste Management System where it would be processed, sampled and monitored prior to release to the environment. Gaseous releases, diluted in the free volume of air in the Nuclear Annex, would be discharged via the Nuclear Annex Ventilation System to the environment. Process radiation monitors in the Nuclear Annex Ventilation System would switch this system to the filtered mode upon detection of radiation. Although, the Nuclear Annex Building Ventilation is not safety-related, realistically the carbon filters in this system would provide an effective means of iodine removal which reduce the dose consequence to the thyroid.

Finally, System 80+™ Technical Specifications, Section 16.7.2, requires the operator to perform a Reactor Coolant System (RCS) water inventory balance every 72 hours for Modes 1, 2, 3, and 4. The System 80+™ Technical Specifications limit the unidentified leakage from the RCS to 1 gpm. The performance of the RCS mass balance would provide an indication of a leak. Once the source of leakage is identified, the operator would isolate the letdown line and terminate the leakage in

RAI 440.105

accordance with System 80+™ Technical Specifications, which requires the operator to reduce the leakage to within the limit in 4 hours. If the leakage can not be reduced to the limit within the required completion time, then the operator must be in mode 3 in 6 hours and in mode 5 in 36 hours. These actions would mitigate the radiological consequences of a small break of the letdown line.

Question 440.115

Technical Specification 3.4.9 of CESSAR-DC Chapter 16 does not include the surveillance requirements for the demonstration of the emergency power supplies for the pressurizer heaters as proposed in the C-E Owners Group Standard Technical Specifications. Explain why.

Response 440.115 (Revision 1)

This surveillance requirement was inadvertently omitted from the System 80+ technical specifications. The response to RAI 430.23 defines pressurizer heater power availability as listed in Table 8.3.1-4.

R1 The pressurizer heater power is supplied from a 4.16KV permanent non-safety bus which may receive emergency power from the non-safety gas turbine or, if necessary, the diesel generator via a manual bus tie. A surveillance requirement will be added to the System 80+ Technical Specifications to demonstrate operability of an emergency power source for pressurizer heaters. This surveillance requirement will be included in a future amendment to Chapter 16.

Question 471.5

Section 12.2.1.1.2.2 states that the average reactor coolant crud activity assumed for CESSAR System 80+ will be maintained within the NUREG-0017 concentrations listed in Table 12.2-6. Discuss how these levels compare with actual measured crud levels at operating CE PWR's and list some of these features incorporated in the CESSAR design that will help to achieve these lower predicted crud levels.

Response 471.5

The radionuclides listed in Table 12.2-6, along with their respective specific activities, are taken from Table 2-2 of NUREG-0017, Rev. 1. These values are those determined to be representative of specific concentrations in a PWR over its lifetime based on data recorded at operating PWR plants.

When actual measured crud levels for operating C-E PWR's are compared to the values in Table 12.2-6, the results are comparable.

In an attempt to reduce crud levels, certain features are incorporated into the System 80+ design.

Materials have been selected with the intent of minimizing the generation and activation of crud in the reactor coolant system. Generally, metallic materials in contact with the reactor coolant are corrosion resistant, such as austenitic stainless steel. Two major sources of crud activity are cobalt and antimony. Crud activity has been minimized by limiting the concentrations of these elements in all surfaces wetted by primary coolant.

Cobalt, present in hardfacing materials such as stellite, is used only when no acceptable substitute material exists. Current studies sponsored by the Electric Power Research Institute (EPRI) are evaluating cobalt-free hardfacing materials for valves; these materials may be available in the near future. Pump bearings use cobalt-free wear materials other than stellite. The other major source of cobalt, impurities in steam generator tubing, has been minimized by limiting cobalt to 0.02 weight percent or less in System 80+ tubing. The steam generators are supplied with Inconel 690 heat transfer tubes, which generally have lower nickel and cobalt contents than Alloy 600, which is currently used. Cobalt content is controlled for all stainless steel wetted surfaces. Wrought austenitic stainless steels and stainless steel cladding is restricted in cobalt content to as low a level as practical.

Antimony is minimized by using antimony-free journal bearings in the reactor coolant pumps. Antimony-free RCP shaft seals are being developed and may be available in the near future. All other pump parts which are wetted by reactor coolant are of antimony-free design.

The chemistry control program for the reactor coolant system also minimizes the generation, transport, and activation of corrosion products. The EPRI primary chemistry guidelines are incorporated in this program. During the pre-core operation period, the RCS is operated at temperatures above 350 F with an elevated pH to form a protective oxide film on metal surfaces. This oxide film resists chemical attack during subsequent plant operations, thereby limiting further corrosion of RCS materials and formation of activated corrosion products. Chemistry control during normal plant operation requires an adequate hydrogen inventory to scavenge corrosion-inducing oxygen. In addition, coordinated lithium-boron control of pH is maintained to minimize the corrosion rate of RCS components as well as the transport of corrosion products. This reduces the precipitation of these corrosion products in the core region. By minimizing corrosion product deposition within the core, the degree of activation of these products is reduced as is the consequent out-of-core radiation fields.

Systems which contain primary coolant are designed and arranged to avoid crud traps.

Question 471-25

Section 12.4.4.B briefly discusses the use of primary system materials having "very low cobalt impurities" for the System 80+ design. Expand this section to describe in more detail the major primary systems components where cobalt has been eliminated or reduced, and provide the maximum cobalt impurity level permitted in System 80+ components in contact with primary system coolant.

Response 471-25

Chapter 12 of CESSAR-DC will be revised to specify the major primary systems components where cobalt has been eliminated or reduced. These changes will appear in a future amendment to CESSAR-DC.

For System 80+ steam generator tubing, a maximum cobalt impurity level of 0.02 weight percent will be permitted. This is consistent with the EPRI ALWR Utility Requirements Document (Chapter 1, Section 5.2.7.1). Cobalt content will also be controlled for all stainless steel wetted surfaces. Wrought astenitic stainless steels and stainless steel cladding will be restricted in cobalt content to as low a level as practical.

the reference plant in 1989. Approximately 3% of the 10,000 personnel received an individual exposure greater than 0.1 rem. Most of these employee were involved in the primary circuit resistance temperature detector (RTD) bypass system replacement or in steam generator maintenance where dose rate fields are in excess of 0.5 rem/hr. The RTD bypass system is not in this CESSAR design.

12.4.4 SYSTEM 80+ UNIQUE ALARA DESIGN FEATURES

This section describes some of the System 80+ design features to achieve ALARA goals.

A. The most successful method of reducing occupational exposure is to eliminate the source of activity. The System 80+ design assures low primary system sources with improved fuel clad leakage performance of less than 0.1% fuel clad failures. This performance is substantially better than past PWR fuel clad leakage based upon historical data. Better fuel performance is expected to reduce total plant exposures by a factor of 1.2 to 1.5

B. Primary System Materials Improvement

Insert A

~~The System 80+ design specifies primary system materials with lower corrosion rates and very low cobalt impurities.~~ I

Steam generator tubes are fabricated to minimize the potential for ~~relieve stresses to reduce~~ stress corrosion cracking. This will reduce the probability of tube plugging activities and further reduce maintenance exposures.

~~Control rod drive materials are specified with low cobalt alloys which in the past have been responsible for a majority of RCS exposures.~~

~~Reduction of cobalt as a source of exposure can reduce total plant exposure by a factor of 1.5 to 2.0~~

C. Reactor Coolant Pump Seals

The System 80+ RCPs incorporate a proven, reliable and easily replaceable seal design. Occupational exposures associated with the seal replacement task are expected to be a factor of 1.5 to 2.0 lower than for other PWR standard designs.

471.25

Insert A:

The System 80+ design specifies primary system materials in contact with primary coolant with low corrosion rates and very low cobalt impurities. A maximum cobalt impurity level of 0.02 weight percent will be specified for System 80+ steam generator tubing. Cobalt content will also be controlled for all stainless steel wetted surfaces. Wrought austenitic stainless steels and stainless steel cladding will be restricted in cobalt content to as low a level as practical.

The use of cobalt alloys and cobalt based hardfacing materials will be minimized. These materials will only be used for applications where no proven alternative exists.

Question 480.8

The CESSAR-DC 80+ system precludes the need to switch-over the emergency core cooling system (ECCS) suction to a separate source of water following the injection phase of a LOCA, by providing an in-containment refueling storage tank (IRWST). However, a single water source, must meet the licensing requirement for the recirculation phase of a LOCA because of the potential buildup of debris and foreign matter. Therefore, provide the following for conformance with GDC 38 and SRP 6.2.2, Revision 4 (NUREG-0800) by following the guidance of RGs 1.82 and 1.1 and NUREG-0897, Revision 1, as appropriate:

a) Perform a Net Positive Suction Head (NPSH) analysis for the containment spray pumps to assure that pump cavitation will not occur during any anticipated operating conditions during the injection and recirculation phases of post LOCA. Also, provide associated supporting inputs and assumptions for the above analysis.

b) Provide an evaluation of the long term ability of the IRWST, to provide a reliable source of water for the containment spray system during the recirculation phase of a LOCA. This evaluation should include provisions for adequate drainage back to the IRWST, IRWST hydraulic performance, and the design features of the IRWST which preclude debris accumulation from inhibiting sufficient flow to the containment spray system.

Response

a) This question has been answered in response to question 480.35a.

b) An evaluation of the IRWST shows that sufficient water exists in the IRWST to maintain CSS pump suction requirements during all phases of a design basis accident (DBA). This evaluation accounts for water held up in containment from containment sprays, and assumes the inadvertent actuation of the reactor cavity (RC) spillway valves. The evaluation shows that sufficient water will accumulate in the hold-up volume tank (HVT) and RC to bring the water level to the IRWST spillway and allow the IRWST to be replenished.

The HVT is sufficiently large to allow heavy debris to settle rather than be transported to the IRWST. Smaller debris will be prevented from entering the suction lines to the SIS, SCS, and CSS by screens at the suction inlets capable of removing particles greater than 0.09 in. diameter

Question 480.35
(6.2.1)

To satisfy the requirements of GDC 16 and 50 regarding sufficient design margin, for plants in the CP stage of review, the containment design pressure should provide at least a 10% margin above the accepted peak calculated containment pressure following a LOCA, or a steam of feedwater line break. The calculated peak pressure found in Table 6.2.1-2 is 48.31 psig and, from Table 6.2.1-3, the internal design pressure is 49.0 psig. Justify the lack of a 10% margin between the peak accident pressure and the design pressure.

Response 480.35
(6.2.1)

Existing plants which underwent a two-stage licensing process (CP and OL stages) have been required by the NRC to provide a 10% margin between the maximum calculated containment pressure (due to LOCA, steam, or feedwater break) and the containment design pressure at the CP stage of their design. The 10% margin requirement is not explicitly defined in either GDC 16 or GDC 50. GDC 50 refers to "sufficient margin" and then describes what input shall be considered when evaluating the available margin. The margin is to be provided to account for uncertainties and conservatisms that may or may not have been considered in the design process. The design process was an ongoing process during the CP stage for all existing plants. Extra margin was needed because the design generally was incomplete at the CP stage of the licensing process. CE's System 80+™ design is one of the first of a new generation of plants which will undergo a single-step licensing process. The design of all essential safety systems will be complete at the time of NRC approval. Therefore, 10% additional margin is not necessary for System 80+™.

The System 80+™ containment is designed for the envelope of a large number of seismic conditions. This arises from the consideration of numerous soil profiles input into the soil-structure-interaction analysis used to develop the response spectra for which the containment is designed. Margin which would normally show in the difference between the calculated and design pressure is used here.

The margin in the pressure values given above (approximately 1.37%) is based on the Level B loading condition which includes the Operating Basis Earthquake (OBE) . The deletion of the OBE from the design requirements of 10CFR100 (it appears the industry and regulators are inclined to move in this direction) would make Level A the controlling loading condition. There is more than 10% margin for loading Levels A, C and D.

Question 480.35 (6.2.2 & 6.5) b.

Provide an evaluation of the long-term performance of the in-containment refueling water storage tank (IRWST) to provide a reliable source of water for the containment spray system during the recirculation phase of a LOCA. This requirement, in accordance with GDC 38 and SRP 6.2.2 Rev. 4, Item II.6, should include an evaluation of adequate drainage back to the IRWST of spray water, IRWST hydraulic performance, and the design features of the IRWST which preclude debris accumulation from inhibiting sufficient flow to the containment spray system. Guidance from Regulatory Guide 1.82 Rev. 1 and NUREG-0897 Rev. 1 should be used in preparing a response.

Response 480.35 (6.2.2 & 6.5)(b)

Adequate drainage is defined as providing for an adequate water inventory to continue circulation. For the purpose of determining the NPSH available, calculations were performed to determine the minimum water level in the IRWST during accident conditions. In the calculation the following conservative assumptions are made:

The only water available is that from the IRWST, i. e., no water is contributed by the reactor coolant system or the safety injection tanks.

With the containment spray (CS) system actuated, the reactor cavity is assumed flooded and the holdup volume tank full such that water levels are equalized and water is returning to the IRWST through the spillways.

Spray water is being held up on surfaces throughout containment.

Locations for the accumulation of water held up inside the containment consist of water held up on horizontal surfaces, clogged floor drains, water held up in containment spray piping, water in the containment atmosphere, water film on vertical surfaces, puddles trapped on equipment, water soaked into insulation, and the containment free volume filled with steam.

This volume of water is conservatively estimated and deducted from the IRWST water inventory. The difference results in an IRWST water volume which corresponds to a water level at elevation of 75+6. This is the minimum draw down elevation achieved before water begins returning to the IRWST.

Based on the preliminary pipe routing and a conservative

RAI 480.35(b)

minimum IRWST water level of 75+6, the available NPSH for the containment spray pumps ranges from 24 feet for the normal CS pump flow rate of 5000 gpm to 21.3 feet for CS pump runout flow of 6500 gpm. This NPSH available is greater than typical containment spray pump NPSH requirements.

In Regulatory Guide 1.82 the recommended hydraulic guidelines for pump air ingestion are based on minimum submergence, maximum Froude number and maximum pipe velocities. These parameters are generated from experimental results using a right rectangular sump model. Since the geometry of the IRWST, as well as the locations of the suction lines, is different from that of the experimental sump models, it is difficult to say whether the empirical data presented in Reg. Guide 1.82 will accurately predict the hydraulic behavior of the IRWST. Nonetheless, assuming the data is valid for the IRWST, and applying the equations generated by the tests to the parameters of the IRWST results in zero air ingestion for normal pump flow and an air ingestion of less than 2% for pump runout using the conservative minimum IRWST water level elevation of 75+6.

It should be noted that during a LOCA, the reactor cavity will not be flooded as was assumed in determining the minimum water level of 75+6. Because of this, an additional volume of water will be available to raise the minimum water level in the IRWST approximately 2 ft. which will further decrease the possibility of air ingestion.

Along with the concern of air ingestion, accumulation of post LOCA debris is a consideration. In the System 80+™ design, debris is prevented from entering the IRWST first by seismically designed vertical screens provided at the entrance to the holdup volume tank at elevation 91+9. In addition, the height of the spillway from the holdup volume to the IRWST will minimize entrance of any debris into the IRWST. Screens are also located at the entrance to the pump suction lines to provide assurance no debris will enter the system.

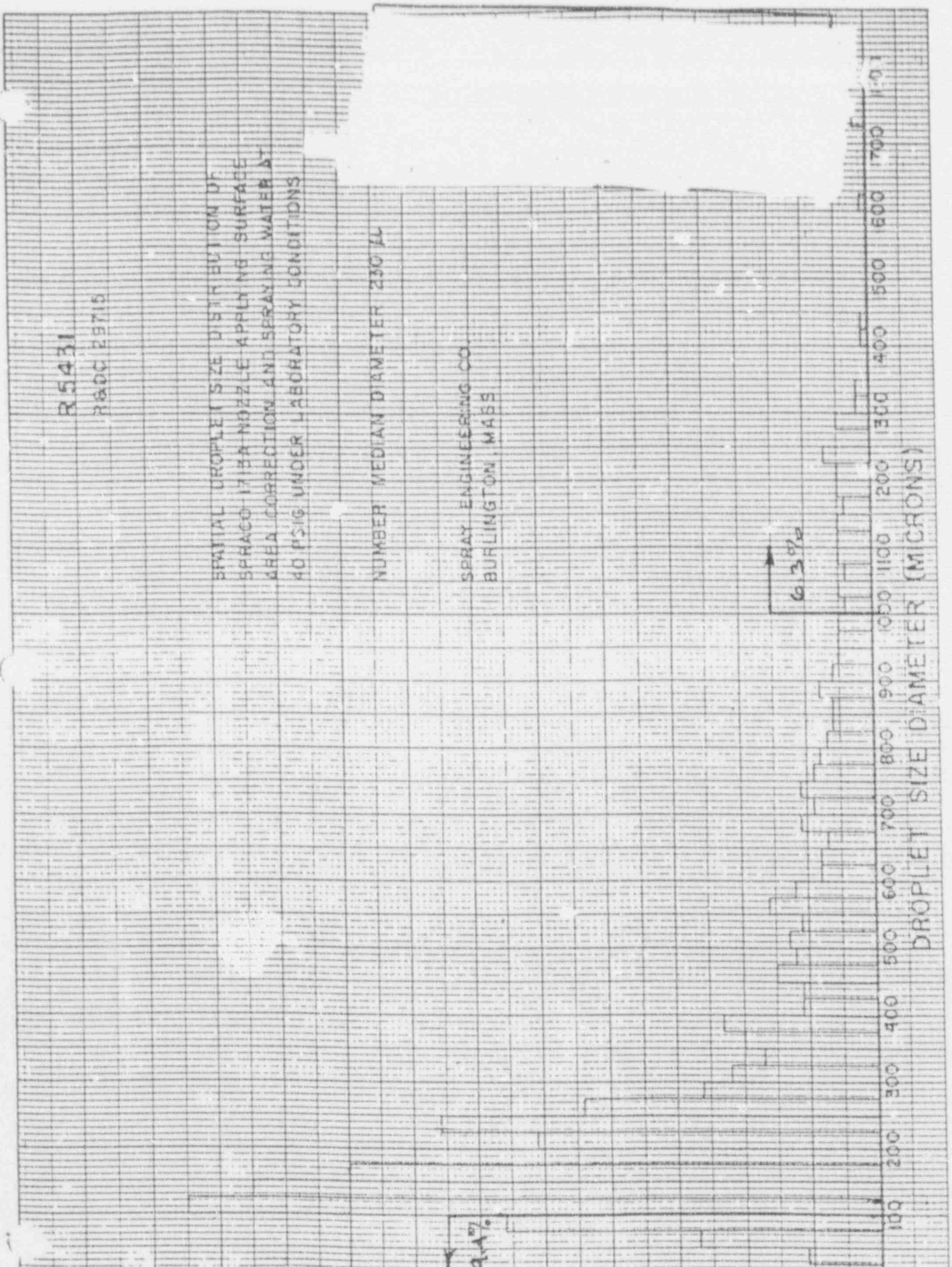
Question 480.35

d. In accordance with SRP 6.5.2 Rev. 1, Item II.1.d, provide detailed information on the drop size distribution for the nozzles, such as a histogram. Designations such as "average," "mean," and "median" numbers do not provide sufficiently detailed information to permit an independent evaluation of the performance of the nozzle.

Response 480.35

d. A histogram is attached giving the spatial drop size distribution of the SPRACO 1713A nozzle as determined by the manufacturer. Spraying conditions utilized in generating the histogram are in accordance with the nozzle design parameters given in Table 6.5-1 of CESSAR-DC.

Attachment ALWR-357 RAI 480.35(d)



Question 500.24(a)

What is the significance of the damage control measures (DCM's) listed in the bold brackets following the description of each scenario in 5.2.B of CESSAR-DC Appendix 13A?

Response 500.24(a)

The bold DCM numbers at the end of each specific scenario for sabotage listed in CESSAR-DC Appendix 13A 5.2.B refer to the Damage Control Measures that would likely be employed to either prevent or mitigate the consequences of the listed sabotage scenario. For example sabotage scenario 5.2.B.4 assumes that both Component Cooling Water pumps are sabotaged rendering them unable to perform their intended function. This event could be mitigated by DCM #16 which provides a cross connect between the Emergency Services Water System and the Component Cooling Water System to provide cooling water to the components normally cooled by CCW.

Question 500.24(b)

Section 5.3 of CESSAR-DC Appendix 13A states that the sabotage evaluation performed "revealed that the System 80+ Standard Design incorporates the necessary damage control measures required to mitigate the scenarios". Please provide details of the evaluation that show the basis for that conclusion.

Response 500.24(b)

The damage control measures were developed by the system design engineers using the guidance provided by:

> Nuclear Power Plant Damage Control Measures and Design Changes for Sabotage Protection NUREG/2585 Sandia National Laboratory, Albuquerque, NM.: Peter Lobner, Science Applications Inc., La Jolla, CA.
>
> Ranking of Light Water Reactor Systems for Sabotage Protection; Lobner P., Goldman L., Horton W., Finn S., Sand 82-7053 Sandia National Laboratory July 1982.
>
> System 80+ Probabilistic Risk Assessment

The assumptions regarding the nature of the saboteur were:

1. Carried no explosives
2. Would disable safety systems or their support systems
3. Would initiate an event or transient
4. Would not take any action to prevent mitigation or recovery of the event by the station staff

The analysis then identified those System 80+ safety systems, and their support systems outside containment, which could be sabotaged and ranked them in order of safety importance as follows:

1. Systems required to shutdown the reactor by addition of negative reactivity
2. Systems required to remove decay heat from RCS
3. Systems required to keep the reactor core at least partially covered with coolant
4. Systems which must be available for immediate operation in the event of an emergency
5. Systems which must be available in a moderate length of time in an emergency

Response 500.24(b) (Cont'd)

6. Systems which would be required after a considerable length of time in an emergency

These rankings are listed in Table 13A.4.2 of Appendix 13A.

The analysis also identified those transients which could be caused by a saboteur acting against equipment outside the containment. Once the vulnerable systems and scenarios were developed, Damage Control Measures were designed into the plant's systems and controls to prevent or mitigate any anticipated sabotage. Each of the DCM's was designed to counteract sabotage using one or more of the strategies listed in Appendix 13A 4.0 (A through D).

Question 500.25(a)

Does the conclusion of Appendix 13A mean that there is no combination of components which, if tampered with, would remain undiscovered long enough to lead to core damage?

Response 500.25(a)

There is no conclusion in Appendix A regarding combinations of components, which if tampered with, would remain undiscovered long enough to lead to core damage. As stated in NUREG 1267, p. 24: "The NRC has sponsored various studies over the years to identify plant vulnerabilities to insider sabotage and to assess the effectiveness of proposed design and procedural modification..."

"The insights from these studies indicate that there were no single design modifications or procedures that would completely eliminate the threat of insider sabotage since, regardless of the design modification, there will be always be access to the system or its installation, maintenance, or operation." However, System 80+ design features such as component and power supply redundancy and separation, and instrumentation separation, and owner/operator measures such as vital area access control, site access screening, fitness for duty programs, and administrative controls all combine to reduce the likelihood of sabotage-initiated events or transients going undetected or unmitigated long enough to lead to core damage. The requirements of 10CFR 73.55(c) to provide adequate physical barriers to protect vital equipment and 10CFR 73.55(d) to identify access control points to all vital areas are incorporated within the System 80+ design. Furthermore the recommendations of NUREG 1267, Technical Resolution of Generic Safety Issue A-29, have been incorporated into the System 80+ design.

Question 500.25(b)

Section 6, item H of Appendix 13A notes that the Nuplex 80+ instrumentation and controls design incorporates "on-line monitoring of fluid and electrical systems making detection of sabotage attempts more likely". Would this system be able to detect mispositioning of manual valves in these systems? If rot discuss timeliness of discovery should a locked valve outside containment be either mistakenly or deliberately mispositioned.

Response 500.25(b)

Manual valves which are necessary for success path fulfillment are instrumented for position. Nuplex 80+ determines whether the current position of the valve supports success path fulfillment and, if not, generates an alarm in the Main Control Room (MCR) and provides operators with the information necessary to correct valve position. In this way detection of, and corrective action for, mispositioned manual valves with direct safety functions can be accomplished.

If the system with the a non-instrumented manual locked valve normally has no flow in it and the valve in question should be locked open but in fact is closed, this condition could persist. If such mispositioning has occurred, and if the system was actuated so as to initiate flow, the Nuplex 80+ instrumentation in the MCR would indicate a lack of flow concurrent with alarms on the MCR annunciator panels. The mispositioned valve could then be opened.

If a non-instrumented normally locked closed manual valve in a system in which there is normally no flow was open, then the presence of flow through the system would be indicated and alarmed in the MCR and appropriate actions taken.

If a non-instrumented normally locked open manual valve in a system in which there is normally flow was closed, then the absence of normal flow would be indicated and alarmed in the MCR and appropriate action taken.

Normal station operating procedures provide check-off lists to preclude inadvertent mispositioning of locked valves. These procedures are used prior to plant and system start up. The use of such check-off lists plus administrative control of keys for locked valves, along with routine and surveillance testing programs, will reduce the likelihood of either inadvertent or deliberate mispositioning of manual valves occurring or going undected.

The Nuplex 80+ design simplifies the detection of mispositioned valves for which the MCR has control and position indication available by flashing the red or green indicating light after valve motion has taken place, thus facilitating awareness of the valve's current and former position.

Response 500.25(b) (Cont'd)

The training of operators in system operations, plant response, administrative procedures and plant layout and component location ensures that should a manually locked valve be mispositioned either inadvertently or by sabotage, abnormal flow conditions will be noticed and prompt action taken. In addition, the use of root cause analysis increases the likelihood that, if an inadvertently mispositioned valve is discovered, the proper corrective actions will be taken to preclude a recurrence or similar event.

Question 500.32

(a) URD Chapter 10, Section 4.9.3.8, includes the statement:

"... any action taken to enable a remote shutdown station or transfer control to it shall be annunciated in the MCR [main control room]. The MMIS [Man-Machine Interface System] Designer shall make an assessment, using analysis and active simulation (including appropriate assumptions regarding support from security personnel), that the annunciation in the MCR provides adequate time for the operator and plant security to take action to prevent a serious accident in the case of unauthorized use of a remote shutdown station."

Does C-E intend to provide such an analysis, and if so, when?

(b) Is there a provision in CESSAR-DC that any action taken to enable a remote shutdown station, or transfer control to it, shall be annunciated in the control room or in the central alarm station.

(c) CESSAR-DC Fig. 9.5.1-3 shows the rooms that are identified in Fig. 1.2-5 as containing the transfer switches to be within a single fire barrier area. Does C-E response to 500.10, that transfer switches are located in four separate equipment rooms each with a single access controlled entrance, still apply? Access to the remote shutdown panel room is shown to be just a few steps from the transfer switch rooms. Would safety considerations permit location of at least two of the transfer switches at a control room exit to ensure that the alarm provides adequate time for the operator and plant security to take actions to prevent a serious accident in the case of unauthorized use of a remote shutdown station.

Response 500.32(a)

(a) Due to the transfer switch geographic locations (in the MCR and channelized equipment rooms), layered security and corresponding alarms provided, unauthorized transfer of control to the Remote Shutdown Panel (RSP) is impractical. Therefore CE did not intend to formally analyze these events per the URD, Chapter 10. Details of the Nuplex 80+ transfer design, including alarms are provided below. Although highly impractical, unauthorized transfer to the RSP (from outside the MCR) can only be accomplished one channel at a time (from separate geographic locations); at least five control channels would remain active in the MCR after the first channel is transferred, giving the operator adequate controls to initiate a safe shutdown. In addition, it is noted that manual reactor trip pushbuttons are never disabled in the MCR. Therefore, trip from the MCR can never be prevented.

500.32

(b) Although CESSAR-DC does not specifically address this feature, the Nuplex 80+ design does include provisions for alarming actions which enable the RSC.

Access to the Remote Shutdown Panel (RSP) is controlled by site security, with an alarm generated in the MCR when the RSP room is accessed. Access to the channelized Instrument and Control (I&C) equipment rooms are also controlled by plant security, with an alarm generated in the MCR when the entry door to an I&C equipment room is opened. The five separate channelized I&C equipment rooms are the only locations outside the MCR from which manual transfer of control can be accomplished (via the Component Control System Maintenance and Test Panels.) Additional alarms are generated in the MCR if a channel transfer is actuated. This combination of alarms ensures that MCR operators are aware of the status of transfer, as well alerting MCR operators and the security force to possible security breaches.

(c) The Nuplex 80+ design for transfer of control from the MCR to the RSP includes a complete set of transfer switches (one per channel) located at each exit from the MCR for transfer as the operator exi[illegible] In addition single channel transfer switches are located in the geographically separated, channelized I&C equipment rooms. The four transfer switch rooms in close proximity to the RSP have been eliminated. The transfer scheme is explained in more detail below. Changes to CESSAR-DC will be submitted in a future amendment. The transfer scheme is similar to the suggested approach provided in question 500.32(c) except transfer of all channels are located in the MCR.

In the event an incident occurs which forces the operator to leave the MCR the operator can transfer control from the MCR to the RSP by activating transfer switches (2 sets, 6 switches each set) located near each MCR exit. The signal from each switch transfers control from the MCR to the RSP and blocks further control system inputs from the MCR, thus isolating spurious signals from fire damage in the MCR. The transfer of control from the MCR to the RSP at the MCR exit is a one-way transfer. Control cannot be transferred back to the MCR from that location. This ensures that any subsequent failures (e.g., due to fire) in the control room in the vicinity of the transfer switches will not reinstate control in the unmanned control room. Transfer of control from the RSP back to the MCR is only accomplished in each of the channelized I&C equipment rooms, through the use of the CCS Maintenance and Test Panel (MTP). The MTP also provides an additional location for single channel transfer capability from the MCR to the RSP.