NRC PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 9 1978

Docket No. STN 50-480

Mr. Tom M. Anderson, Manager
Nuclear Safety Department
Westinghouse Electric Corporation
P.O. Box 355
Nuclear Center - Bay 415
Pittsburgh, Pennsylvania 15230

Dear Mr. Anderson:

SUBJECT: EXTENSION REVIEW MATTERS FOR PRELIMINARY DESIGN APPROVALS

The Commission's August 22, 1978 policy statement on standardization includes a provision which allows any Preliminary Design Approval (PDA) that had been previously issued for a three-year term to be extended for two additional years. This provision applies to PDA-3 issued for the RESAR-41 application.

As set forth in the policy statement, each application for a PDA extension will be subject to an assessment of the design with respect to the Category I, II, III, and IV matters approved for implementation since the regulatory requirements cutoff date for the PDA in question. A tabulation of each Category I, II, III, and IV matter approved since October 23, 1974, the regulatory requirements cutoff date for RESAR-41, is provided in Enclosures A, B, C, and D, respectively, to this letter.

Should you desire PDA-3 to be extended for two additional years, we request that you provide an assessment of the RESAR-41 design against each Category I, II, III, and IV matter identified in the enclosures which is applicable to the RESAR-41 design. Upon receipt of your responses, the staff will review them as follows:

1) The staff will review your responses to determine whether they are complete. If the staff determines that your responses are complete, we will administratively extend PDA-3 for two additional years subject to later staff acceptance of your proposed resolution for the applicable Category II, III, and IV matters identified in the enclosures.

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2) If the RESAR-41 design is to be referenced in a construction permit application, we will initiate a detailed extension review as soon as we are informed by a utility-applicant that it will reference the RESAR-41 design. It is anticipated that the staff will require additional information in order to complete the detailed extension review. You will be notified of the additional information requirements as soon as they are determined. The staff's detailed extension review will be conducted according to the following guidelines:

   a) Category I Matters - This review will determine whether you have clearly delineated the extent to which the design already conforms to these matters. It is anticipated that there will be no changes to the design resulting from the staff's review of Category I matters.

   b) Category II Matters - This review will define the extent to which the design conforms, or provides an acceptable alternative, to these matters. For those cases where the design is not in substantial conformance with these matters or acceptable alternatives are not provided you should demonstrate why conformance is not necessary. The outcome of the staff review may result in additional requirements.

   c) Category III Matters - This review will determine the extent to which the design conforms to these matters or whether acceptable alternatives are provided. If the design does not conform to the stated Category III requirements or no acceptable alternative has been provided, staff-approved revisions to the design will be required.

   d) Category IV Matters - Category IV matters are those which have not been reviewed by the RRRC, but which the Director, NRR, deems to have sufficient safety attributes to warrant their being addressed during the PDA extension review. These matters will be treated identically to the Category II matters.

Your response to each matter identified in the enclosures should be submitted as an amendment to the RESAR-41 application in the form of a new appendix. Changes should not be made to the main body of the RESAR-41 Safety Analysis Report at this time.

The RESAR-41 Safety Analysis Report appendix addressing the extension review matters should be filed on the PDA docket prior to December 15, 1978. If a Final Design Approval (FDA) application for the RESAR-41 design is tendered the PDA extension matters may also be filed and reviewed on the FDA docket. Review on the FDA docket is acceptable if (1) the extension review matters can be resolved on a schedule consistent with the review schedule established or contemplated for any construction permit application referencing the RESAR-41 PDA, and (2) the construction permit applicant(s) agree with that course of action.

The matter of whether licensing fees will be assessed for the PDA extension reviews was not addressed in the Commission's August 22, 1978 policy statement on standardization. The Commission has this matter under consideration. We will advise you of the Commission's decision on fees for PDA extensions as soon as it becomes available.

If you require any clarification of the matters discussed in this letter, please contact Patrick D. O'Reilly, the staff's assigned licensing project manager.

Sincerely,

RLC for

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosures:
A. Category I Matters
B. Category II Matters
C. Category III Matters
D. Category IV Matters
E. Description of Other Positions Identified as NRR Category IV Matters in Enclosure D

ENCLOSURE A

CATEGORY I MATTERS APPROVED BY RRRC

	EFFECTIVE DATE	DOCUMENT NO.	REVISION	TITLE
	1/31/78	RG 1.7	2	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident
	9/1/78	RG 1.9	1	Selection, Design, and Qualification for Diesel-Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants
	1/9/76	RG 1.20	2	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing
	11/29/77	RG 1.28	1	Quality Assurance Program Requirements (Design and Construction)
	6/20/78	RG 1.29	3	Seismic Design Classification
	7/20/76	RG 1.31	2	Control of Ferrite Content in Stainless Steel Weld Metal
	1/14/77	RG 1.32	2	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants
	10/21/76	RG 1.33	1	Quality Assurance Program Requirements (Operation)
	8/15/75	RG 1.35	2	Inservice Inspection of Ungrouted Tendons in Pre-stressed Concrete Containment Structures
(1)	5/77	RG 1.38	2	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants
	7/12/77	RG 1.39	2	Housekeeping Requirements for Water-Cooled Nuclear Power Plants

	EFFECTIVE DATE	DOCUMENT NO.	REVISION	TITLE
(2)	11/29/77	RG 1.52	2	Design, Testing, and Maintenance for Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants
(3)	3/22/77	RG 1.63	1	Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants
	1/9/76	RG 1.64	2	Quality Assurance Requirements for the Design of Nuclear Power Plants
	6/20/78	RG 1.68	2	Initial Test Programs for Water-Cooled Reactor Power Plants
	9/26/75	RG 1.68.1	0	Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants
	11/15/77	RG 1.72	1	Spray Pond Plastic Piping
(1)	3/78	RG 1.84	12	Code Case Acceptability - ASME Section III Design and Fabrication
(1)	3/78	RG 1.85	12	Code Case Acceptability - ASME Section III Materials
	5/26/77	RG 1.90	1	Inservice Inspection of Pre-stressed Concrete Containment Structures with Grouted Tendons
	8/22/75	RG 1.92	1	Combining Modal Responses and Spatial Components in Seismic Response Analysis
	2/6/76	RG 1.94	1	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel during the Construction Phase of Nuclear Power Plants
	10/21/76	RG 1.95	1	Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release

	EFFECTIVE DATE	DOCUMENT NO.	REVISION	TITLE
(4)	1/14/77	RG 1.99	1	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials
	6/14/77	RG 1.100	1	Seismic Qualification of Electric Equipment for Nuclear Power Plants
(1)	10/76	RG 1.103	1	Post-Tensioned Pre-stressing Systems for Concrete Reactor Vessels and Containments
	1/28/77	RG 1.106	1	Thermal Overload Protection for Electric Motors on Motor-Operated Valves
	10/21/76	RG 1.107	1	Qualifications for Cement Grouting for Pre-stressing Tendons in Containment Structures
(1)	5/77	RG 1.116	0-R	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems
	9/27/77	RG 1.118	1	Periodic Testing of Electric Power and Protection Systems
(5)	5/11/77	RG 1.120	1	Fire Protection Guidelines for Nuclear Power Plants
	11/15/77	RG 1.122	1	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components
(1)	7/77	RG 1.123	1	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants

	EFFECTIVE DATE	DOCUMENT NO.	REVISION	TITLE
	1/14/77	RG 1.126	0	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification
	6/20/78	RG 1.128	1	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants
	2/18/77	RG 1.129	0	Maintenance, Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants
(6)	5/26/77	RG 1.131	0	Qualification Tests of Electric Cables, Field Splices and Connections for Light Water Cooled Nuclear Power Plants
	5/11/77	RG 1.132	0	Site Investigations for Foundations of Nuclear Power Plants
	3/22/77	RG 1.134	0	Medical Certification and Monitoring of Personnel Requiring Operator Licenses
	7/12/77	RG 1.135	0	Normal Water Level and Discharge at Nuclear Power Plants
	8/31/77	RG 1.136	0	Material for Concrete Containments
(7)	9/27/77	RG 1.137	0	Fuel Oil Systems for Standby Diesel Generators
	9/27/77	NUREG-0102 (SRP 1.8)	0	Interfaces for Standard Designs
	11/15/77	RG 1.138	0	Laboratory Investigation of Soils for Engineering Analysis and Design of Nuclear Power Plants
	11/15/77	RG 1.XXX	0	Permanent Dewatering Systems
	11/29/77	RG 1.140	0	Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of LWR's
	1/31/78	RG 1.142	0	Safety-Related Concrete Structures
	3/14/78	RG 8.19	0	Occupational Radiation Dose Assessment at LWR's - Design Stage Man-Rem Estimates

	EFFECTIVE DATE	DOCUMENT NO.	REVISION	TITLE
(8)	3/14/78	RSB 5-2	0	Reactor Coolant System Overpressure Protection

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(1) Indicates that the category I assigned by RRRC for the previous revision of this document was retained. Review by the RRRC for reassignment of the category is not required for document revisions which do not result in an increase in requirements.

(2) Revision 1 of this regulatory guide was assigned as a Category II matter effective January 9, 1976. It is the intent of the RRRC that revision 1 remain a Category II matter. However, revision 2 may be used in lieu of revision 1 if so desired by applicants.

(3) Assigned as a Category II matter by the RRRC for those applications not previously reviewed to revision 0.

(4) Category I for paragraph C.3 only. Paragraphs C.1, C.2, and C.4 are assigned by the RRRC as Category III matters.

(5) In specifying category I for this regulatory guide, the RRRC recognizes that the staff is utilizing Appendix A to BTP ASB 9.5-1 on operating reactors and all CP and OL applications now under review.

(6) In specifying the category I for this regulatory guide, the RRRC recognizes that the fire protection aspects are covered by Appendix A to BTP ASB 9.5-1 which is a Category II matter

(7) Category I for all CP or PDA applications docketed after the implementation date shown in the published guide. Certain provisions of the guide are also assigned by the RRRC as Category II and Category III matters.

(8) Category I for operating licenses issued prior to March 14, 1978. Assigned by the RRRC as a Category III matter for all other applications.

ENCLOSURE B

CATEGORY II MATTERS APPROVED BY RRRC

	EFFECTIVE DATE	DOCUMENT	REVISION	TITLE
	11/12/75	RG 1.27	2	Ultimate Heat Sink for Nuclear Power Plants
	1/9/76	RG 1.52	1	Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants
(1)	8/77	RG 1.59	2	Design Basis Floods for Nuclear Power Plants
(2)	3/22/77	RG 1.63	1	Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants
(3)	5/16/78	RG 1.68.2	1	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants
	11/15/77	RG 1.91	1	Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites
	1/28/77	RG 1.97	1	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident
	11/12/75	RG 1.102	1	Flood Protection for Nuclear Power Plants
	9/15/76	RG 1.105	1	Instrument Setpoints
	6/14/77	RG 1.108	1	Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants

	EFFECTIVE DATE	DOCUMENT	REVISION	TITLE
	3/22/77	RG 1.115	1	Protection Against Low-Trajectory Turbine Missiles
	12/20/77	RG 1.117	1	Tornado Design Classification
	8/31/77	RG 1.124	1	Service Limits and Loading Combinations for Class 1 Linear Type Component Supports
	7/77	RG 1.130	0	Design Limits and Loading Combinations for Class 1 Plate- and Shell-Type Component Supports
(4)	9/29/77	RG 1.137	0	Fuel Oil Systems for Standby Diesel Generators
	8/18/76	RG 8.8	2	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low As is Reasonably Achievable (Nuclear Power Reactors)
	8/18/76	BTP ASB 9.5-1		Guidelines for Fire Protection for Nuclear Power Plants Under Review and Construction
	4/13/77	BTP MTEB 5-7		Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping
(5)	1/31/78	SRP 5.4.7	1	Residual Heat Removal System
(6)	1/31/78	RG 1.141	0	Containment Isolation Provisions for Fluid Systems

(1) Indicates that the category II assigned by RRRC for the previous revision of this document was retained. Review by the RRRC for reassignment of the category is not required for document revisions which do not result in an increase in requirements.

(2) Assigned as a Category I matter for those applications previously reviewed to revision 0. Category II for all other applications.

(3) Category II for operating reactors. Assigned by the RRRC as a Category III matter for all other applications.

(4) Category II for paragraph C.1 for all CP's or PDA's under review whose SER's have not been issued prior to the implementation date shown in the published guide. Paragraph C.2 for all operating reactors, OL applications, and CP and PDA applications under review whose SER's are completed prior to the implementation date shown in the published guide. Certain provisions of this guide are also assigned by the RRRC as Category III matters.

(5) Category II for operating reactors and all other applications for which the issuance of the OL is expected prior to January 1, 1979. Assigned by the RRRC as a Category III matter for all other applications.

(6) Category II for operating reactors and OL reviews. Assigned by the RRRC as a Category III matter for all other applications.

ENCLOSURE C

CATEGORY III MATTERS APPROVED BY RRRC

	EFFECTIVE DATE	DOCUMENT	REVISION	TITLE
	5/16/78	RG 1.56	1	Maintenance of Water Purity in Boiling Water Reactors
(1)	5/16/78	RG 1.68.2	1	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants
(2)	1/14/77	RG 1.99	1	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials
(3)	3/77	RG 1.101	1	Emergency Planning for Nuclear Power Plants
	11/76	RG 1.114	1	Guidance on Being Operator at the Controls of a Nuclear Power Plant
	5/11/76	RG 1.121	0	Bases for Plugging Degraded PWR Steam Generator Tubes
	11/29/77	RG 1.127	1	Inspection of Water-Control Structures Associated with Nuclear Power Plants
(4)	9/27/77	RG 1.137	0	Fuel Oil Systems for Standby Diesel Generators
(5)	1/31/78	SRP 5.4.7	1	Residual Heat Removal System
(6)	1/31/78	RG 1.141	0	Containment Isolation Provisions for Fluid Systems
(7)	3/14/78	RSB 5-2	0	Reactor Coolant System Overpressurization Protection

ENCLOSURE C

CATEGORY III MATTERS APPROVED BY RRRC FROM MARCH 1974 THROUGH AUGUST 1978

	EFFECTIVE DATE	DOCUMENT	REVISION	TITLE
	5/16/78	RG 1.56	1	Maintenance of Water Purity in Boiling Water Reactors
(1)	5/16/78	RG 1.68.2	1	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants
(2)	1/14/77	RG 1.99	1	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials
(3)	3/77	RG 1.101	1	Emergency Planning for Nuclear Power Plants
	11/76	RG 1.114	1	Guidance on Being Operator at the Controls of a Nuclear Power Plant
	5/11/76	RG 1.121	0	Bases for Plugging Degraded PWR Steam Generator Tubes
	11/29/77	RG 1.127	1	Inspection of Water-Control Structures Associated with Nuclear Power Plants
(4)	9/27/77	RG 1.137	0	Fuel Oil Systems for Standby Diesel Generators
(5)	1/31/78	SRP 5.4.7	1	Residual Heat Removal System
(6)	1/31/78	RG 1.141	0	Containment Isolation Provisions for Fluid Systems
(7)	3/14/78	RSB 5-2	0	Reactor Coolant System Overpressurization Protection

(1) Assigned by RRRC as a Category II matter for operating plants.

(2) Paragraph C.3 is a Category I matter.

(3) Indicates that the Category III assigned by RRRC for the previous revision of this document was retained. Review by the RRRC for reassignment of the category is not required for document revisions which do not result in an increase in requirements.

(4) Category III for paragraph C.2 for all CP and PDA applications under review whose SER's have not been issued prior to the implementation date shown in the published guide. Certain provisions of this guide are also assigned by the RRRC as Category II matters.

(5) Category III for CP or PDA applications docketed prior to January 1, 1978, and for which OL issuance is expected after January 1, 1979, all Category II for all other applications.

(6) Assigned by RRRC as a Category II matter for operating reactors and OL applications.

(7) Assigned by RRRC as a Category I matter for OL's issued prior to March 14, 1978, and Category III for all other applications.

ENCLOSURE D

NRR CATEGORY IV MATTERS

A. Regulatory Guides not categorized

	Issue Date	Number	Revision	Title
1.	12/75	1.13	1	Spent Fuel Storage Facility Design Basis
2.	8/75	1.14	1	Reactor Coolant Pump Flywheel Integrity
3.	1/75	1.75	1	Physical Independence of Electric Systems
4.	9/75	1.79	1	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors
5.	7/75	1.83	1	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes
6.	11/74	1.89	0	Qualification of Class 1E Equipment for Nuclear Power Plants
7.	12/74	1.93	0	Availability of Electric Power Sources
8.	2/76	1.104	0	Overhead Crane Handling Systems for Nuclear Power Plants

B. SRP Criteria

Implementation Date	Branch	Applicable SRP Section	Title
1. 11/24/75	MTEB	5.4.2.1	BTP MTEB-5-3, Monitoring of Secondary Side Water Chemistry in PWR Steam Generators
2. 11/24/75	CSB	6.2.1 6.2.1A 6.2.1B 6.2.1.2 6.2.1.3 6.2.1.4 6.2.1.5	BTP CSB-6-1, Minimum Containment Pressure Model for PWR ECCS Performance Evaluation
3. 11/24/75	CSB	6.2.5	BTP CSB-6-2, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident
4. 11/24/75	CSB	6.2.3	BTP CSB-6-3, Determination of Bypass Leakage Path in Dual Containment Plants
5. 11/24/75	CSB	6.2.4	BTP CSB-6-4, Containment Purging During Normal Plant Operations
6. 11/24/75	ASB	9.1.4	BTP ASB-9.1, Overhead Handling Systems for Nuclear Power Plants
7. 11/24/75	ASB	10.4.9	BTP ASB-10.1, Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for PWR's
8. 11/24/75	SEB	3.5.3	Procedures for Composite Section Local Damage Prediction (SRP Section 3.5.3, par. II.1.C)

Implementation Date	Branch	Applicable SRP Section	Title
9. 11/24/75	SEB	3.7.1	Development of Design Time History for Soil-Structure Interaction Analysis (SRP Section 3.7.1, par. II.2)
10. 11/24/75	SEB	3.7.2	Procedures for Seismic System Analysis (SRP Section 3.7.2 par. II)
11. 11/24/75	SEB	3.7.3	Procedures for Seismic Subsystem Analysis (SRP Section 3.7.3, par. II)
12. 11/24/75	SEB	3.8.1	Design and Construction of Concrete Containments) SRP Section 3.8.1, par. II)
13. 11/24/75	SEB	3.8.2	Design and Construction of Steel Containments (SRP Section 3.8.2, par. II)
14. 11/24/75	SEB	3.8.3	Structural Design Criteria for Category I Structures Inside Containment (SRP Section 3.8.3, par. II)
15. 11/24/75	SEB	3.8.4	Structural Design Criteria for Other Seismic Category I Structures. (SRP Section 3.8.4, par. II)
16. 11/24/75	SEB	3.8.5	Structural Design Criteria for Foundations (SRP Section 3.8.5, par. II)
17. 11/24/75	SEB	3.7 11.2 11.3 11.4	Seismic Design Requirements for Radwaste Sysems and Their Housing Structures (SRP Section 11.2, BTP ETSB 11-1,par. B.v)

	Implementation Date	Branch	Applicable SRP Section	Title
18.	11/24/75	SEB	3.3.2	Tornado Load Effect Combinations (SRP Section 3.3.2, par. II.2.d)
19.	11/24/75	SEB	3.4.2	Dynamic Efects of Wave Action (SRP Section 3.4.2, par. II)
20.	10/01/75	ASB	10.4.7	Water Hammer for Steam Generators with Preheaters (SRP Section 10.4.7 par. I.2.b)
21.	11/24/75	AB	4.4	Thermal-Hydraulic Stability (SRP Section 4.4, par. II.5)
22.	11/24/75	RSB	5.2.5	Intersystem Leakage Detection (SRP Section 5.2.5 par. II.4) and R.G. 1.45
23.	11/24/75	RSB	3.2.2	Main Steam Isolation Valve Leakage Control System (SRP Section 10.3 par. III.3 and BTP RSB-3.2)

C. Other Positions

	Implementation Date	Branch	Applicable SRP Section	Title
1.	12/1/76	SEB	3.5.3	Ductility of Reinforced Concrete and Steel Structural Elements Subjected to Impactive or Impulsive Loads
2.	8/01/76	SEB	3.7.1	Response Spectra in Vertical Direction
3.	4/01/76	SEB	3.8.1 3.8.2	BWR Mark III Containment Pool Dynamics
4.	9/01/76	SEB	3.8.4	Air Blast Loads
5.	10/01/76	SEB	3.5.3	Tornado Missile Impact
6.	6/01/77	RSB	6.3	Passive Failures During Long-Term Cooling Following LOCA

Implementation Date	Branch	Applicable SRP Section	Title
7. 9/01/77	RSB	6.3	Control Room Position Indication of Manual (Handwheel) Valves in the ECCS
8. 4/01/77	RSB	15.1.5	Long-Term Recovery from Steamline Break: Operator Action to Prevent Overpressurization
9. 12/01/77	RSB	5.4.6 5.4.7 6.3	Pump Operability Requirements
10. 3/28/78	RSB	3.5.1	Gravity Missiles, Vessel Seal Ring Missiles Inside Containment
11. 1/01/77	AB	4.4	Core Thermal-Hydraulic Analysis
12. 1/01/78	PSB	8.3	Degraded Grid Voltage Conditions
13. 6/01/76	CSB	6.2.1.2	Asymmetric Loads on Components Located Within Containment Subcompartments
14. 9/01/77	CSB	6.2.6	Containment Leak Testing Program
15. 1/01/77	CSB	6.2.1.4	Containment Response Due to Main Steam Line Break and Failure of MSLIV to Close
16. 11/01/77	ASB	3.6.1 3.6.2	Main Steam and Feedwater Pipe Failures
17. 1/01/77	ASB	9.2.2	Design Requirements for Cooling Water to Reactor Coolant Pumps
18. 8/01/76	ASB	10.4.7	Design Guidelines for Water Hammer in Steam Generators with Top Feedring Design (BTP ASB-10.2)
19. 1/01/76	ICSB	3.11	Environmental Control Systems for Safety Related Equipment

ENCLOSURE E

DESCRIPTION OF OTHER POSITIONS IDENTIFIED AS NRR CATEGORY IV MATTERS IN ENCLOSURE D

Numbering scheme corresponds to that used in Item C of Enclosure D; e.g., the first "Other Position" identified as a Category IV matter in Item C of Enclosure D is designated IV.C.1, etc.

ENCLOSURE E

IV.C.1 (3.5.3) DUCTILITY OF REINFORCED CONCRETE AND STEEL STRUCTURAL ELEMENTS SUBJECTED TO IMPACTIVE OR IMPULSIVE LOADS

INTRODUCTION

In the evaluation of overall response of reinforced concrete structural elements (e.g., missile barriers, columns, slabs, etc.) subjected to impactive or impulsive loads, such as impacts due to missiles, assumption of non-linear response (i.e., ductility ratios greater than unity) of the structural elements is generally acceptable provided that the safety functions of the structural elements and those of safety-related systems and components supported or protected by the elements are maintained. The following summarizes specific SEB interim positions for review and acceptance of ductility ratios for reinforced concrete and steel structural elements subjected to impactive and impulsive loads.

SPECIFIC POSITIONS

1. REINFORCED CONCRETE MEMBERS

1.1 For beams, slabs, and walls where flexure controls design, the permissible ductility ratio (μ) under impactive and impulsive loads should be taken as

$$\mu = \frac{0.05}{\rho - \rho'} \quad \text{for} \quad \rho - \rho' \geq .005$$

$$\mu = 10 \quad \text{for} \quad \rho - \rho' \leq .005$$

where ρ and ρ' are the ratios of tensile and compressive reinforcing as defined in ACI-318-71 Code.

1.2 If use of a ductility ratio greater than 10 (i.e., $\mu > 100$) is required to demonstrate design adequacy of structural elements against impactive or impulsive loads, e.g., missile impact, such a usage should be identified in the plant SAR. Information justifying the use of this relatively high ductility value shall be provided for SEB staff review.

1.3 For beam-columns, walls, and slabs carrying axial compression loads and subject to impulsive or impactive loads producing flexure, the permissible ductility ratio in flexure should be as follows:

(a) When compression controls the design, as defined by an interaction diagram, the permissible ductility ratio shall be 1.3.

(b) When the compression loads do not exceed $0.1f_c{}'Ag$ or one-third of that which would produce balanced conditions, whichever is smaller, the permissible ductility ratio can be as given in Section 1.1.

(c) The permissible dutility ratio shall vary linearly from 1.3 to that given in Section 1.1 for conditions between those specified in (a) and (b). (See Fig 1.)

1.4 For structural elements resisting axial compressive impulsive or impactive loads only, without flexure, the permissible axial ductility ratio shall be 1.3.

1.5 For shear carried by concrete only

$\mu = 1.0$

For shear carried by concrete and stirrups or bent bars

$\mu = 1.3$

For shear carried entirely by stirrups

$\mu = 3.0$

2.0 STRUCTURAL STEEL MEMBERS

2.1 For flexure compression and shear

$\mu = 10.0$

2.2 For columns with slenderness ratio (l/r) equal to or less than 20

$\mu = 1.3$

where l = effective length of the member
r = the least radius of gyration

For columns with slenderness ratio greater than 20

$\mu = 1.0$

2.3 For members subjected to tension

$$\mu = .5 \frac{\varepsilon_U}{\varepsilon_Y}$$

where ε_U = uniform ultimate strain of the material

ε_Y = strain at yield of material

IV.C.2 (3.7.1) RESPONSE SPECTRA IN THE VERTICAL DIRECTION

Subsequent to the issuance of Regulatory Guide 1.60, the report "Statistical Studies of Vertical and Horizontal Earthquake Spectra" was issued in January 1976 by NRC as NUREG-0003. One of the important conclusions of this report is that the response spectrum for vertical motion can be taken as 2/3 the response spectrum for horizontal motion over the entire range of frequencies in the Western United States. According to Regulatory Guide 1.60, the vertical response spectrum is equal to the horizontal response spectrum between 3.5 cps and 33 cps. For the Western United States only, consistent with the latest available data in NUREG-0003, the option of taking the vertical design design response spectrum as 2/3 the horizontal response spectrum over the entire range of frequencies will be accepted. For other locations, the vertical response spectrum will be the same as that given in Regulatory Guide 1.6C.

IV.C.3 (3.8.1 3.8.2) BWR MARK III CONTAINMENT POOL DYNAMICS

1. POOL SWELL

a. Bubble pressure, bulk swell and froth swell loads, drag pressure and other pool swell loads should be treated as abnormal pressure loads, P_a. Appropriate load combinations and load factors should be applied accordingly.

b. The pool swell loads and accident pressure may be combined in accordance with their actual time histories of occurrence.

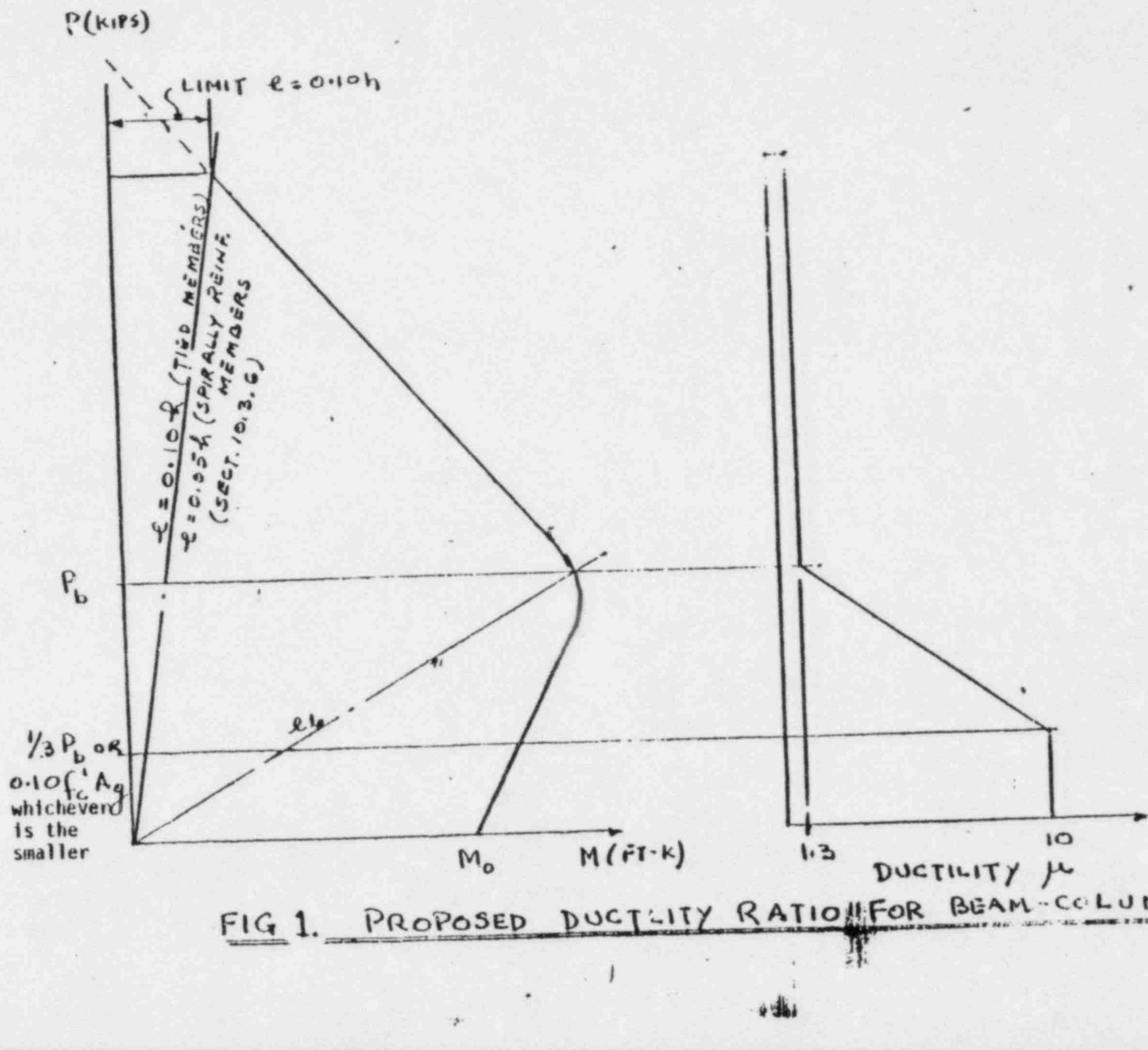

FIG 1. PROPOSED DUCTILITY RATIO FOR BEAM-COLUMNS

2. SAFETY RELIEF VALVE (SRV) DISCHARGE

   a. The SRV loads should be treated as live loads in all load combinations $1.5P_a$ where a load factor of 1.25 should be applied to the appropriate SRV loads.

   b. A single active failure causing one SRV discharge must be considered in combination with the Design Basis Accident (DBA).

   c. Appropriate multiple SRV discharge should be considered in combination with the Small Break Accident (SBA) and Intermediate Break Accident (IBA).

   d. Thermal loads due to SRV discharge should be treated as T_o for normal operation and T_a for accident conditions.

   e. The suppression pool liner should be designed in accordance with the ASME Boiler and Pressure Vessel Code, Division 1 Subsection NE to resist the SRV negative pressure, considering strength, buckling and low cycle fatigue.

IV.C.4 (3.8.4) AIR BLAST LOADS (Pa, Ta, To as defined in ACI 359-740)

The following interim position on air blast loadings on Nuclear Power Plant Structures should be used as guidance in evaluating analyses.

1. An equivalent static pressure may be used for structural analysis purposes. The equivalent static pressure should be obtained from the air blast reflected pressure or the overpressure by multiplying these pressures by a factor of two. Any proposed use of a dynamic load factor less than two should be treated on a case by case basis. Whether the reflected pressure or the overpressure is to be used for individual structural elements depends on whether an incident blast wave could strike the surface of the element.

2. No load factor need be specified for the air blast loads, and the load combination should be:

   $U = D + L + B$

   where, U is the strength capacity of a section
   D is dead load
   L is live load
   B is air blast load.

3. Elastic analysis for air blast is required for concrete structures of new plants. For steel structural elements, and also for reinforced concrete elements in existing plants, some inelastic response may be permitted with appropriate limits on ductility ratios.

4. Air blast generated ground shock and air blast wind pressure may be ignored. Air blast generated missiles may be important in situations where explosions are postulated to occur in vessels which may fragment.

5. Overturning and sliding stability should be assessed by multiplying the structure's full projected area by the equivalent static pressure and assuming only the blast side of the structure is loaded. Justification for reducing the average equivalent static pressure on curved surfaces should be considered on a case by case basis.

6. Internal supporting structures should also be analyzed for the effects of air blast to determine their ability to carry loads applied directly to exterior panels and slabs. Moreover.in vented structures, interior structures may require analysis even if they do not support exterior structures.

7. The equivalent static pressure should be considered as potentially acting both inward and outward.

IV.C.5 (3.5.3) TORNADO MISSILE PROTECTION

As an interim measure, the minimum concrete wall and roof thickness for tornado missile protection (based on the acceptable tornado missile spectra identified in Section 3.5.1.4 of the Standard Review Plan) will be as follows:

	Concrete Strength (psi)	Wall Thickness (inches)	Roof Thickness (inches)
Region I	3000	27	24
	4000	24	21
	5000	21	18
Region II	3000	24	21
	4000	21	18
	5000	19	16
Region III	3000	21	18
	4000	18	16
	5000	16	14

These thicknesses are for protection against local effects only. Designers must establish independently the thickness requirements for overall structura response. Reinforcing steel should satisfy the provisions of Appendix C, ACI 349 (that is, .2% minimum, EWEF). The regions are described in Regulatory Guide 1.76.

IV.C.6 (6.3) PASSIVE ECCS FAILURES DURING LONG-TERM COOLING FOLLOWING A LOCA

Passive failures ' the ECCS, having leak rates equal to or less than those from the suc len failure of a pump seal and which may occur during the long-term cooling period following a postulated LOCA, should be considered. To mitigate the effects of such leaks, a leak detection system having design features and bases as described below should be included in the plant design.

The leak detection system should include detectors and alarms which would alert the operator of passive ECCS leaks in sufficient time so that appropriate diagnostic and corrective actions may be taken on a timely basis. The diagnostic and corrective actions would include the identification and isolation of the faulted ECCS line before the performance of more than one subsystem is degraded. The design bases of the leak detection system should include:

(1) Identification and justification of the maximum leak rate;

(2) Maximum allowable time for operator action and justification therefor;

(3) Demostration that the leak detection system is sensitive enough to initiate and alarm on a timely basis, i.e., with sufficient lead time to allow the operator to identify and isolate the faulted line before the leak can create undesireable consequences such as flooding of redundant equipment. The minimum time to be considered is 30 minutes;

(4) Demonstration that the leak detection system can identify the faulted ECCS train and that the leak can be isolated; and

(5) Alarms that conform with the criteria specified for the control room alarms and a leak detection system that conforms with the requirements of IEEE-279, except that the single failure criterion need not be imposed.

IV.C.7 (6.3) CONTROL ROOM POSITION INDICATION OF MANUAL (HANDWHEEL) VALVES

Regulatory Guide 1.47 specifies automatic position indication of each bypass or deliberately induced inoperable condition if the following three conditions are met:

(1) The bypass or inoperable condition affects a system that is designed to perform an automatic safety function.

(2) The bypass or inoperable condition can reasonably be expected to occur more frequently than once per year.

(3) The bypass or inoperable condition is expected to occur when the system is normally required to operate.

Revision one of the Standard Review Plan in Section 6.3 requires conformance with Regulatory Guide 1.47 with the intent being that any manual (handwheel) valve which could jeopardize the operation of the ECCS, if inadvertently left in the wrong position, must have position indication in the control room. In the PDA extension reviews it is important to confirm that standard designs include this design feature. Most standard designs do but this matter was probably not specifically addressed in some of the first PDA reviews.

IV.C.8 (15.1.5) LONG-TERM RECOVERY FROM STEAM LINE BREAK - OPERATOR ACTION TO PREVENT OVERPRESSURIZATION (PWR)

A steam line break causes cooldown of the primary system, shrinkage of RCS inventory and depletion of pressurizer fluid. Subsequent to plant trip, ECCS actuation, and main steam system isolation, the RCS inventory increases and expands, refilling the pressurizer. Without operator action, replenishment of RCS inventory by the ECCS and expansion at low temperature could repressurize the reactor to an unacceptable pressure-temperature region thereby compromising reactor vessel integrity. Analyses are required to show that following a main steam line break that (i) no additional fuel failures result from the accident, and (ii) the pressures following the initiation of the break will not compromise the integrity of the reactor coolant pressure boundary giving due consideration to the changes in coolant and material temperatures. The analyses should be based on the assumption that operator action will not be taken until ten minutes after initiation of the ECCS.

IV.C.9 (5.4.6 5.4.7 6.3) PUMP OPERABILITY REQUIREMENTS

In some reviews, the staff has found reasonable doubt that some types of engineered safety feature pumps would continue to perform their safety function in the long term following an accident. In such instances there has been followup, including pump redesign in some cases, to assure that long term performance could be met. The following kinds of information may be sought on a case-by-case basis where such doubt arises.

a. Describe the tests performed to demonstrate that the pumps are capable of operating for extended periods under post-LOCA conditions, including the effects of debris. Discuss the damage to pump seals caused by debris over an extended period of operation.

b. Provide detailed diagrams of all water cooled seals and components in the pumps.

c. Provide a description of the composition of the pump shaft seals and the shafts. Provide an evaluation of loss of shaft seals.

d. Discuss how debris and post-LOCA environmental conditions were factored into the specifications and design of the pump.

IV.C.10 GRAVITY MISSILES, VESSEL SEAL RING MISSILES INSIDE CONTAINMENT

(3.5.1) Safety related systems should be protected against loss of function due to internal missiles from sources such as those associated with pressurized components and rotating equipment. Such sources would include but not be limited to retaining bolts, control rod drive assemblies, the vessel seal ring, valve bonnets, and valve stems. A description of the methods used to afford protection against such potential missiles, including the bases therefor, should be provided (e.g., preferential orientation of the potential missile sources, missile barriers, physical separation of redundant safety systems and components). An analysis of the effects of such potential missiles on safety related systems, including metastably supported equipment which could fall upon impingement, should also be provided.

IV.C.11 CORE THERMAL-HYDRAULIC ANALYSES
(4.4)

In evaluating the thermal-hydraulic performance of the reactor core, the following additional areas should be addressed:

1. The effect of radial pressure gradients at the exit of open lattice cores.

2. The effect of radial pressure gradients in the upper plenum.

3. The effect of fuel rod bowing.

In addition, a commitment to perform tests to verify the transient analysis methods and codes is required.

IV.C.12 DEGRADED GRID VOLTAGE CONDITIONS
(8.3)

As a result of the Millstone Unit Number 2 low grid voltage occurrence, the staff has developed additional requirements concerning (a) sustained degraded voltage conditions at the offsite power source, and (b) interaction of the offsite and onsite emergency power systems. These additional requirements are defined in the following staff position.

1. We require that a second level of voltage protection for the onsite power system be provided and that this second level of voltage protection satisfy the following requirements:

   a) The selection of voltage and time set points shall be determined from an analysis of the voltage requirements of the safety-related loads at all onsite system distribution levels;

   b) The voltage protection shall include coincidence logic to preclude spurious trips of the offsite power source;

c) The time delay selected shall be based on the following conditions:

(i) The allowable time delay, including margin, shall not exceed the maximum time delay that is assumed in the SAR accident analyses;

(ii) The time delay shall minimize the effect of short duration disturbances from reducing the availability of the offsite power source(s); and

(iii) The allowable time duration of a degraded voltage condition at all distribution system levels shall not result in failure of safety systems or components;

(iv) The voltage sensors shall automatically initiate the disconnection of offsite power sources whenever the voltage set point and time delay limits have been exceeded;

(v) The voltage sensors shall be designed to satisfy the applicable requirements of IEEE Std. 279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations"; and

(vi) The Technical Specifications shall include limiting conditions for operation, surveillance requirements, trip set points with minimum and maximum limits, and allowable values for the second-level voltage protection sensors and associated time delay devices.

2. We require that the system design automatically prevent load shedding of the emergency buses once the onsite sources are supplying power to all sequenced loads on the emergency buses. The design shall also include the capability of the load shedding feature to be automatically reinstated if the onsite source supply breakers are tripped. The automatic bypass and reinstatement feature shall be verified during the periodic testing identified in Item 3 of this position.

3. We require that the Technical Specifications include a test requirement to demonstrate the full functional operability and independence of the onsite power sources at least once per 18 months during shutdown. The Technical Specifications shall include a requirement for tests: (a) simulating loss of offsite power; (b) simulating loss of offsite power in conjunction with a safety injection actuation signal; and (c) simulating interruption and subsequent reconnection of onsite power sources to their respective buses.

4. The voltage levels at the safety-related buses should be optimized for the full load and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power source by appropriate adjustment of the voltage tap settings of the intervening transformers. We require that the adequacy of the design in this regard be verified by actual measurement, and by correlation of measured values with analysis results.

IV.C.13 ASYMMETRIC LOADS ON COMPONENTS
(6.2.1.2) LOCATED WITHIN CONTAINMENT SUBCOMPARTMENTS

In the unlikely event of a pipe rupture inside a major component subcompartment, the initial blowdown transient would lead to pressure loadings on both the structure and the enclosed component(s). The staff's generic Category A Task Action Plan A-2 is designed to develop generic resolutions for this matter. Our present schedule calls for completing A-2 for PWR's during the first quarter, 1979. Pending completion of A-2, the staff is implementing the following program:

1. For PWRs at the CP/PDA stage of review, the staff requires applicants to commit to address the safety issue as part of their application for an operating license.

2. For PWRs at the OL/FDA stage of review, the staff requires case-by-case analyses, including implementation of any indicated corrective measusres prior to the issuance of an operating license.

3. For BWRs, for which this issue is expected to be of lesser safety significance, the asymmetric loading conditions will be evaluated on a case-specific basis prior to the issuance of an operating license.

For those cases which analyses are required, we request the performance of a subcompartment, multi-node pressure response analysis of the pressure transient resulting from postulated hot-leg and cold-leg (pump suction and discharge) reactor coolant system pipe ruptures within the reactor cavity, pipe penetrations, and steam generator compartments. Provide similar analyses for the pressurizer surge and spray lines, and other high energy lines located in containment compartments that may be subject to pressurization. Show how the results of these analyses are used in the design of structures and component supports.

IV.C.14 (6.2.6) CONTAINMENT LEAK TESTING PROGRAM

To avoid difficulties experienced in this area in recent OL reviews, the staff has increased its scope of inquiry at the CP/PDA stage of review. For this purpose, the following information with regard to the containment leak testing program should be supplied.

a. Those systems that will remain fluid filled for the Type A test should be identified and justification given.

b. Show the design provisions that will permit the personnel air-lock door seals and the entire air lock to be tested.

c. For each penetration,i.e., fluid system piping, instrument, electrical, and equipment and personnel access penerations, identify the Type B and/or Type C local leak testing that will be done.

d. Verify that containment penetrations fitted with expansion bellows will be tested at P_a. Identify any penetration fitted with expansion bellows that does not have the design capability for Type B testing and provide justification.

IV.C.15 (6.2.1.4) CONTAINMENT RESPONSE DUE TO MAIN STEAM LINE BREAK AND MSLIV FAILURE

In recent CP and OL application reviews, the results of analyses for a postulated main steam line break accident (MSLB) for designs utilizing pressurized water reactors with conventional containments show that the peak calculated containment temperature can exceed for a short time period the environmental qualification temperature-time envelope for safety related instruments and components. This matter was also discussed in Issue No. 1 of NUREG-0138 and Issue No. 25 of NUREG-0153. The signifiance of the matter is that it could result in a requirement for requalifying safety-related equipment to higher time-temperature envelopes.

The staff's generic Category A Task Action Plans A-21 and A-24 are designed to develop generic resolutions for these matters. The presently scheduled completion dates for A-21 and A-24 (Short Term Portion) are first quarter, 1979 and fourth quarter, 1978, respectively. Pending completion of A-21 and A-24, some interim guidance will be used as detailed below.

We have developed and are implementing a plan in which all applicants for construction permits and operating licenses and those already issued construction permits must provide information to establish a conservative temperature-time envelope.

Therefore, describe and justify the analytical model used to conservatively determine the maximum containment temperature and pressure for a spectrum of postulated main steam line breaks for various reactor power levels. Include the following in the discussion.

(1) Provide single active failure analyses which specifically identify those safety grade systems and components relied upon to limit the mass and energy release and containment pressure/ temperature response. The single failure analyses should include, but not necessarily be limited to: main steam and connected systems isolation; feedwater auxiliary feedwater, and connected systems isolation; feedwater, condensate, and auxiliary feedwater pump trip, and auxiliary feedwater run-out control system; the loss of or availability of offsite power; diesel failure when loss of offsite power is evaluated; and partial loss of containment cooling systems.

(2) Discuss and justify the assumptions made regarding the time at which active containment heat removal systems become effective.

(3) Discuss and justify the heat transfer correlation(s) (e.g., Tagami, Uchida) used to calculate the heat transfer from the containment atmosphere to the passive heat sinks, and provide a plot of the heat transfer coefficient versus time for the most severe steam line break accident analyzed.

(4) Specify and justify the temperature used in the calculation of condensing heat transfer to the passive heat sinks; i.e., specify whether the saturation temperature corresponding to the partial pressure of vapor, or the atmosphere temperature (which may be superheated) was used.

(5) Discuss and justify the analytical model including the thermodynamic equations used to account for the removal of the condensed mass from the containment atmosphere due to condensing heat transfer to the passive heat sinks;

(6) Provide a table of the peak values of containment atmosphere temperature and pressure for the spectrum of break areas and power levels analyzed;

(7) For the case which results in the maximum containment atmosphere temperature, graphically show the containment atmosphere temperature, the containment liner temperature, and the containment concrete temperature as a function of time. Compare the calculated containment atmosphere temperature response to the temperature profile used in the environmental qualification program for those safety related instruments and mechanical components needed to mitigate the consequences of the assumed main steam line break and effect safe reactor shutdown;

(8) For the case which results in maximum containment atmosphere pressure, graphically show the containment pressure as a function of time; and

(9) For the case which results in the maximum containment atmosphere pressure and temperature, provide the mass and energy release data in tabular form.

In order to demonstrate that safety-related equipment has been adequately qualified as described above, provide the following information regarding its environmental qualification.

(1) Provide a comprehensive list of equipment required to be operational in the event of a main steamline break (MSLB) accident. The list should include, but not necessarily be limited to, the following safety related equipment:

(a) Electrical containment penetrations;

(b) Pressure transmitters;

(c) Containment isolation valves;

(d) Electrical power cables;

(e) Electrical instrumentation cable; and

(f) Level transmitters.

Describe the qualification testing that was, or will be, done on this equipment. Include a discussion of the test environment, namely, the temperature, pressure, moisture content, and chemical spray, as a function of time.

(2) It is our position that the thermal analysis of safety related equipment which may be exposed to the containment atmosphere following a main steam line break accident should be based on the following:

(a) A condensing heat transfer coefficient based on the recommendations in Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation,"should be used.

(b) A convective heat transfer coefficient should be used when the condensing heat flux is calculated to be less than the convective heat flux. During the blowdown period it is appropriate to use a conservatively evaluated forced convection heat transfer correlation. For example,

$$Nu = C(Re)$$

Where Nu = Nusselt No.

Re = Reynolds No.

C = empirical constants dependent on geometry and Reynolds No.

Since the Reynolds number is dependent on velocity, it is necessary to evaluate the forced flow currents which will be generated by the steam generaor blowdown. The CVTR experiments provide limited data in this regard. Convective currents of from 10 ft/sec to 30 ft/sec were measured locally. We recommend that the CVTR test results be extrapolated conservatively to obtain forced flow currents to determine the convective heat transfer coefficient during the blowdown period. After the blowdown has ceased or been reduced to a negligibly low value, a natural convection heat transfer correlation is acceptable.

(3) For each component where thermal analysis is done in conjunction with an environmental test at a temperature lower than the peak calculated temperature following a main steam line break accident, compare the test thermal response of the component with the accident thermal analysis of the component. Provide the basis by which the component thermal response was developed from the environmental qualification test program. For instance, graphically show the thermocouple data and discuss the thermocouple locations, method of attachment, and performance characteristics, or provide a detailed discussion of the analytical model used to evaluate the component thermal response during the test. This evaluation should be performed for the potential points of failure such as thin cross-sections and temperature sensitive parts where thermal stressing, temperature-related degradation, steam or chemical interaction at elevated temperatures, or other thermal effects could result in the failure of the component mechanically or electrically. If the component thermal response comparison results in the prediction of a more severe thermal transient for the accident conditions than for the qualification test, provide justification that the affected component will perform its intended function during a MSLB accident, or provide protection for the component whch would appropriately limit the thermal effects.

IV.C.16 ENVIRONMENTAL EFFECT OF PIPE FAILURES
(3.6.1,
3.6.2) Identify the "break exclusion" regions of the main steam and feedwater lines. Compartments that contain break exclusion regions of main steam and feedwater lines and any safety related equipment in these compartments should be designed to withstand the environmental effects (pressure, temperature, humidity and flooding) of a crack with a break area equal to the cross sectional area of the "break excluded" pipe.

IV.C.17 DESIGN REQUIREMENTS FOR COOLING WATER
(5.4.1) TO REACTOR COOLANT PUMPS

Demonstrate that the reactor coolant system (RCS) pump seal injection flow will be automatically maintained for all transients and accidents or that enough time and information are available to permit corrective action by an operator.

We have established the following criteria for that portion of the component cooling water (CCW) system which interfaces with the reactor coolant pumps to supply cooling water to pump seals and bearings during normal operation, anticipated transients, and accidents.

1. A single active failure in the component cooling water system shall not result in fuel damage or a breach of the reactor coolant pressure boundary (RCPB) caused by an extended loss of cooling to one or more pumps. Single active failures include operator error, spurious actuation of motor-operated valves, and loss of CCW pumps.

2. A pipe crack or other accident (unanticipated occurrence) shall not result in either a breach of the RCPB or excessive fuel damage when an extended loss of cooling to two or more RC pumps occurs. A single active falure shall be considered when evaluating the consequences of this accident. Moderate leakage cracks should be determined in accordance with Branch Technical Position ASB 3-1.

In order to meet the criteria established above, an NSSS interface requirement should be imposed on the balance-of-plant CCW system that provides cooling water to the RC pump seals and motor and pump bearings, so that the system will meet the following conditions:

1. That portion of the component cooling water (CCW) system which supplies cooling water to the reactor coolant pumps and motors may be designed to non-seismic Category I requirements and Quality Group D if it can be demonstrated that the reactor coolant pumps will operate without component cooling water for at least 30 minutes without loss of function or the need for operator protective action. In addition, safety grade instrumentation including alarms should be provided to detect the loss of component cooling water to the reactor coolant pumps and motors, and to notify the operator in the control room. The entire instrumentation system, including audible and visual alarms, should meet the requirements of IEEE Std 279-1971.

If it is not demonstrated that the reactor coolant pumps and motors will operate at least 30 minutes without loss of function or operator protective action, then the design of the CCW system must meet the following requirements:

1. Safety grade instrumentation consistent with the criteria for the reactor protection system shall be provided to initiate automatic protection of the plant. For this case, the component cooling water supply to the seals and pump and motor bearings may be designed to non-seismic Category I requirements and Quality Group D; or

2. The component cooling water supply to the pumps and motors shall be capable of withstanding a single active failure or a moderate energy line crack as defined in our Branch Technical Position APCSB 3-1 and be designed to seismic Category I, Quality Group D and ASME Section III, Class 3 requirements.

The reactor coolant (RC) pumps and motors are within the NSSS scope of design. Therefore, in order to demonstrate that an RC pump design can operate with loss of component cooling water for at least 30 minutes without loss of function or the need for operator action, the following must be provided:

1. A detailed description of the events following the loss of component cooling water to the RC pumps and an analysis demonstrating that no consequences important to safety may result from this event. Include a discussion of the effect that the loss of cooling water to the seal coolers has on the RC pump seals. Show that the loss of cooling water does not result in a LOCA due to seal failure.

2. A detailed analysis to show that loss of cooling water to the RC pumps and motors will not cause a loss of the flow coastdown characteristics or cause seizure of the pumps, assuming no administrative action is taken. The response should include a detailed description of the calculation procedure including:

   a. The equations used.

   b. The parameters used in the equations, such as the design parameters for the motor bearings, motor, pump and any other equipment entering into the calculations, and material property values for the oil and metal parts.

   c. A discussion of the effects of possible variations in part dimensions and material properties, such as bearing clearance tolerances and misalignment.

   d. A description of the cooling and lubricating systems (with appropriate figures) associated with the RC pump and motor and their design criteria and standards.

   e. Information to verify the applicability of the equations and material properties chosen for the analysis (i.e., references should be listed, and if empirical relations are used, provide a comparison of their range of application to the range used in the analysis).

   Should an analysis be provided to demonstrate that loss of component cooling water to the RC pumps and motor assembly is acceptable, we will require certain modifications to the plant Technical Specifications and an RC pump test conducted under operating condtions and with component cooling water terminated for a specified period of time to verify the analysis.

IV.C.18 (10.4.7) WATER HAMMER IN STEAM GENERATORS WITH TOP FEEDRING DESIGN

Events such as damage to the feedwater system piping at Indian Point Unit No. 2, November 13, 1973, and at other plants, could originate as a consequence of uncovering of the feedwater sparger in the steam generator or uncovering of the steam generator feedwater inlet nozzles. Subsequent events may in turn lead to the generation of a pressure wave that is propagated through the pipes and could result in unacceptable damage.

For CP/PDA and OL/FDA applications, provide the following for steam generators utilizing top feed:

1. Prevent or delay water draining from the feedring following a drop in steam generator water level by means such as J-Tubes;

2. Minimize the volume of feedwater piping external to the steam generator whch could pocket steam using the shortest possible (less than seven feet) horizontal run of inlet piping to the steam generator feedring; and

3. Perform tests acceptable to the staff to verify that unacceptable feed-water hammer will not occur using the plant operating procedures for normal and emergency restoration of steam generator water level following loss of normal feedwater and possible draining of the feedring. Provide the procedures for these tests for staff approval before conducting the tests.

Furthermore, we request that the following be provided:

a. Describe normal operating occurrences of transients that could cause the water level in the steam generator to drop below the sparger or nozzles to cause uncovering and allow steam to enter the sparger and feedwater piping.

b. Describe your criteria or show by isometric diagrams, the routing of the feedwater piping from the steam generators outwards to beyond the containment structure up to the outer isolation valve and restraint.

c. Describe any analysis on the piping system including any forcing functions that will be performed or the results of test programs to verify that,either uncovering of feedwater lines could not occur or that, if it did occur, unacceptable damage such as the experience at the Indian Point Unit No. 2 facility would not result with your design.

IV.C.19 (3.11) ENVIRONMENTAL CONTROL SYSTEMS FOR SAFETY RELATED EQUIPMENT

Most plant areas that contain safety related equipment depend on the continuous operation of environmental control systems to maintain the environment in those areas within the range of environmental qualification of the safety related equipment installed in those areas. It appears that there are no requirements for maintaining these environmental control systems in operation while the plant is shutdown or in hot standby conditions. During periods when these environmental control systems are shutdown, the safety related equipment could be exposed to environmental conditions for which it has not been qualified. Therefore, the safety related equipment should be qualified to the extreme environmental conditions that could occur when the control equipment is shutdown or these environmental control systems should operate continuously to maintain the environmental conditions within the qualification limits of the safety related equipment. In the second case an environmental monitoring system that will alarm when the environmental conditions exceed those for which safety related equipment is qualified shall be provided. This environmental monitoring system shall (1) be of high quality, (2) be periodically tested and calibrated to verify its continued functioning, (3) be energized from continuous power sources, and (4) provide a continuous record of the environmental parameters during the time the environmental conditions exceed the normal limits.