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

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Mr. L. C. Dail, Vice President
Design Engineering Department
Duke Power Company
P. O. Box 33189
Charlotte, North Carolina 28242

Dear Mr. Dail:

SUBJECT: INFORMATION RELATING TO CATEGORIZATION OF RECENT REGULATORY GUIDES BY THE REGULATORY REQUIREMENTS REVIEW COMMITTEE - PERKINS NUCLEAR STATION

We have recently advised utilities with plants in the post-CP phase of the reactor licensing process of the status of NRC staff review and use of recently-approved regulatory guides, and have indicated how these guides would be used in the Operating License review of their Final Safety Analysis Reports. Such information, while not directly applicable to you at this time, may nonetheless be useful to you for your future planning. The text of our letter to these utilities is the following:

"SUBJECT: IMPLEMENTATION OF STAFF REVIEW REQUIREMENTS - (Name of Plant) - OPERATING LICENSE REVIEW

During the last several years, we have reviewed and approved several new regulatory guides and branch technical positions or other modifications to existing staff positions. Our practice is that substantive changes in staff positions be considered by the NRC's Regulatory Requirements Review Committee (RRRC) which then recommends a course of action to the Director, Office of Nuclear Reactor Regulation (NRR). The recommended action includes an implementation schedule. The Director's approval then is used by the NRR staff as review guidance on individual licensing matters. Some of these actions will affect your application. This letter is intended to bring you up to date on these changes in staff positions so that you may consider them in your Final Safety Analysis Report (FSAR) preparation.

TA4
CCP

DPM:LWR #4	DPM:LWR #4	DPM:LWR
CMoon:tlb	SVarga	DVassallo
1/25/79	1/26/79	1/26/79

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"The RRRC applies a categorization nomenclature to each of its actions. (A copy of the summary of RRRC Meeting No. 31 concerning this categorization is attached as Enclosure 1.) Category 1 matters aare those to be applied to applications in accordance with the implementation section of the published guide. We have enclosed lists of actions which are either Category 2 or Category 3, which are defined as follows:

Category 2: A new position whose applicability is to be determined on a case-by-case basis. You should describe the extent to which your design conforms, or you should describe an acceptable alternate, or you should demonstrate why conformance is not necessary.

Category 3: Conformance or an acceptable alternative is required. If you do not conform, or do not have an acceptable alternate, then staff-approved design revisions will be required.

"We believe that providing you with a list of the Category 2 and 3 matters approved to date will be useful in your FSAR preparation, and they will be an essential part of our operating license review. Enclosure 2 is a list of the Category 2 matters. Enclosure 3 is a list of the Catetory 3 matters.

"In addition to the RRRC categories, there also exists an NRR Category 4 list which are those matters not yet reviewed by the RRRC, but which the Director, NRR, has deemed to have sufficient attributes to warrant their being addressed and considered in ongoing reviews. These matters will be treated like Category 2 matters until such time as they are reviewed by the RRRC, and a definite implementation program is developed. A current list of Category 4 matters is attached (Enclosure 4). These also should be considered in your FSAR.

"In some instances the items in the enclosures may not be applicable to your application. Also, we recognize that your application may, in some instances, already conform to the stated staff positions. In your FSAR you should note such compliance.

"If you have any questions please let us know."

For your information, I am enclosing a set of the enclosures that accompanied these individual letters. These enclosures list the present Category 1-4 matters discussed in the letter.

Sincerely,

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

Enclosures:
As stated

cc: See next page

Duke Power Company

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UNITED STATES
NUCLEAR REGULATORY COMMISSI.
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SEP 2 4 1975

Lee V. Gossick
Executive Director for Operations

REGULATORY REQUIREMENTS REVIEW COMMITTEE MEETING NO. 31, JULY 11, 1975

1. The Committee discussed issues related to the implementation of Regulatory Guides on existing plants and the concerns expressed in the June 24, 1974 memorandum, A. Giambusso to E. G. Case, subject: REGULATORY GUIDE IMPLEMENTATION, and made the following recommendations and observations:

   a. Approval of new Regulatory Guides and approval of revisions of existing guides should move forward expeditiously in order that the provisions of these regulatory guides be available for use as soon as possible in on-going or future staff reviews of license applications. The Committee noted that over the recent past, the approval of proposed regulatory guides whose content is acceptable for these purposes has experienced significant delays in RRRC review pending the determination of the applicability of the guide to existing plants, often requiring significant staff effort. To avoid these delays, the Committee concluded that, henceforth, approval of proposed regulatory guides should be uncoupled from the consideration of their backfit applicability.

   b. The implementation section of new regulatory guides should address, in general, only the applicability of the guide to applications in the licensing review process using, in so far as possible, a standard approach of applying the guide to those applications docketed 8 months after the issuance date of the guide for comment. Exceptions to this general approach will be handled on a case-by-case basis.

   c. The regulatory position of each approved proposed guide (or proposed guide revision) will be characterized by the Committee as to its backfitting potential, by placing it in one of three categories:

      Category 1 - Clearly forward fit only. No further staff consideration of possible backfitting is required.

ENCLOSURE 1

<u>Category 2</u> - Further staff consideration of the need for backfitting appears to be required for certain identified items of the regulatory position--these individual issues are such that existing plants need to be evaluated to determine their status with regard to these safety issues in order to determine the need for backfitting.

<u>Category 3</u> - Clearly backfit. Existing plants should be evaluated to determine whether identified items of the regulatory position are resolved in accordance with the guide or by some equivalent alternative.

From time to time, for a specific guide, there will probably be some variation among these categories or even within a category, and these three broad category characterizations will be qualified as required to meet a particular situation.

d. It is not intended that the Committee categorization appear in the guide itself. The purpose of the categorization is to indicate those items of the regulatory position for which the Committee can make a specific backfit recommendation without additional staff work (Categories 1 and 3), and to indicate those items for which additional staff work is required in order to determine backfit considerations (Category 2).

e. The Committee recommends that for approved guides in Category 2, staff efforts be initiated in parallel with the process leading to publication of the guide in order that specific backfit requirements for existing plants be determined within a reasonable period of time after publication of the guide.

f. The Committee observed that more attention needs to be given to the identification of acceptable alternatives to the positions outlined in the guides in order to provide additional options and flexibility to applicants and licensees, with the possible benefits of additional innovation and exploration in the solution of safety issues.

2. The Committee reviewed the proposed Regulatory Guide 1.XX: THERMAL OVERLOAD PROTECTION FOR MOTORS ON MOTOR-OPERATED VALVES and recommended approval. This guide was characterized by the Committee as Category 1 - no backfitting, with the stipulation that as an appropriate occasion presented itself in conjunction with the review of some particular aspect of existing plants, the thermal overload protection provisions be audited.

3. The Committee reviewed the proposed Regulatory Guide 1.XX: INSTRUMENT SPANS AND SETPOINTS and recommended approval subject to the following comment:

   Paragraph 5 of Section C (page 4 of the proposed Guide) should be reworded in light of Committee comments, to the satisfaction of the Director, Office of Standards Development. This guide was characterized by the Committee as Category 1 - no backfit.

4. The Committee reviewed Proposed Regulatory Guide 1.97: INSTRUMENTATION FOR LIGHT WATER COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT CONDITIONS DURING AND FOLLOWING AN ACCIDENT and deferred further consideration to a later meeting in order to permit incorporation of recent comments by the Division of Technical Review.

Edson G. Case, Chairman
Regulatory Requirements Review
Committee

CATEGORY 2 MATTERS

Document Number	Revision	Date	Title
RG 1.27	2	1/76	Ultimate Heat Sink for Nuclear Power Plants
RG 1.52	1	7/76	Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants (Revision 2 has been published but the changes from Revision 1 to Revision 2 may, but need not, be considered.
RG 1.59	2	8/77	Design Basis Floods for Nuclear Power Plants
RG 1.63	2	7/78	Electric Penetration Assemblies in Containment Structures for Light Water Cooled Nuclear Power Plants
RG 1.91	1	2/78	Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites
RG 1.102	1	9/76	Flood Protection for Nuclear Power Plants
RG 1.105	1	11/76	Instrument Setpoints
RG 1.108	1	8/77	Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants
RG 1.115	1	7/77	Protection Against Low-Trajectory Turbine Missiles
RG 1.117	1	4/78	Tornado Design Classification
RG 1.124	1	1/78	Service Limits and Loading Combinations for Class 1 Linear Type Component Supports
RG 1.130	0	7/77	Design Limits and Loading Combinations for Class 1 Plate- and Shell-Type Component Supports

(Continued)

Continued

Document Number	Revision	Date	Title
RG 1.137	0	1/78	Fuel Oil Systems for Standby Diesel Generators (Paragraph C.2)
RG 8.8	2	3/77	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will be as Low as is Reasonably Achievable (Nuclear Power Reactors)
BTP ASB 9.5-1	1		Guidelines for Fire Protection for Nuclear Power Plants (See Implementation Section, Section D)
BTP MTEB 5-7		4/77	Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping
RG 1.141	0	4/78	Containment Isolation Provisions for Fluid Systems

September 15, 1978

CATEGORY 3 MATTERS

Document Number	Revision	Date	Title
RG 1.99	1	4/77	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials (Paragraphs C.1 and C.2.
RG 1.101	1	3/77	Emergency Planning for Nuclear Power Plants
RG 1.114	1	11/76	Guidance on Being Operator at the Controls of a Nuclear Power Plant
RG 1.121	0	8/76	Bases for Plugging Degraded PWR Steam Generator Tubes
RG 1.127	1	3/78	Inspection of Water-Control Structures Associated with Nuclear Power Plants
RSB 5-1	1	1/78	Branch Technical Position: Design Requirements of the Residual Heat Removal System
RSB 5-2	0	3/78	Branch Technical Position: Reactor Coolant System Overpressurization Protection (Draft copy attached)
RG 1.97	1	8/77	Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident (Paragraph C.3 - with additional guidance on paragraph C.3.d to be provided later)
RG 1.68.2	1	7/78	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants
RG 1.56	1	7/78	Maintenance of Water Purity in Boiling Water Reactors

Attachment:
BTP RSB 5-2 (Draft)

DRAFT

BRANCH TECHNICAL POSITION RSB 5-2

OVERPRESSURIZATION PROTECTION OF PRESSURIZED WATER REACTORS

WHILE OPERATING AT LOW TEMPERATURES

A. Background

General Design Criterion 15 of Appendix A, 10 CFR 50, requires that "the Reactor Coolant System and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

Anticipated operational occurrences, as defined in Appendix A of 10 CFR 50, are "those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power."

Appendix G of 10 CFR 50 provides the fracture toughness requirements for reactor pressure vessels under all conditions. To assure that the Appendix G limits of the reactor coolant pressure boundary are not exceeded during any anticipated operational occurrences, Technical Specification pressure-temperature limits are provided for operating the plant.

The primary concern of this position is that during startup and shutdown conditions at low temperature, especially in a water-solid condition, the reactor coolant system pressure might exceed the reactor vessel pressure-temperature limitations in the Technical Specifications established for protection against brittle fracture. This inadvertent overpressurization could be generated by any one of a variety of malfunctions or operator errors. Many incidents have occurred in operating plants as described in Reference 1.

Additional discussion on the background of this position is contained in Reference 1.

DRAFT

B. Branch Position

1. A system should be designed and installed which will prevent exceeding the applicable Technical Specifications and Appendix G limits for the reactor coolant system while operation at low temperatures. The system should be capable of relieving pressure during all anticipated overpressurization events at a rate sufficient to satisfy the Technical Specification limits, particularly while the reactor coolant system is in a water-solid condition.

2. The system must be able to perform its function assuming any single active component failure. Analyses using appropriate calculational techniques must be provided which demonstrate that the system will provide the required pressure relief capacity assuming the most limiting single active failure. The cause for initiation of the event, e.g., operator error, component malfunction, will not be considered as the single active failure. The analysis should assume the most limiting allowable operating conditions and systems configuration at the time of the postulated cause of the overpressure event. All potential overpressurization events must be considered when establishing the worst case event. Some events may be prevented by protective interlocks or by locking out power. These events should be reviewed on an individual basis. If the interlock/power lockout is acceptable, it can be excluded from the analyses provided the controls to prevent the event are in the plant Technical Specifications.

3. The system must meet the design requirements of IEEE 279 (see Implementation). The system may be manually enabled, however, the electrical instrumentation and control system must provide alarms to alert the operator to:

   a. properly enable the system at the correct plant condition during cooldown,

   b. indicate if a pressure transient is occurring.

4. To assure operational readiness, the overpressure protection system must be tested in the following manner:

   a. A test must be performed to assure operability of the system electronics prior to each shutdown.

   b. A test for valve operability must, as a minimum be conducted as specified in the ASME Code Section XI.

   c. Subsequent to system, valve, or electronics maintenance, a test on that portion(s) of the system must be performed prior to declaring the system operational.

5. The system must meet the requirements of Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants" and Section III of the ASME Code.

6. The overpressure protection system must be designed to function during an Operating Basis Earthquake. It must not compromise the design criteria of any other safety-grade system with which it would interface, such that the requirements of Regulatory Guide 1.29, "Seismic Design Classification" are met.

7. The overpressure protection system must not depend on the availability of offsite power to perform its function.

8. Overpressure protection systems which take credit for an active component(s) to mitigate the consequences of an overpressurization event must include additional analyses considering inadvertent system initiation/actuation or provide justification to show that existing analyses bound such an event.

C. <u>Implementation</u>

The Branch Technical Position, as specified in Section B, will be used in the review of all Preliminary Design Approval (PDA), Final Design Approval (FDA), Manufacturing License (ML), Operating License (OL), and Construction Permit (CP) applications involving plant designs incorporating pressurized water reactors. All aspects of the position will be applicable to all applications, including CP applications utilizing the replication option of the Commission's standardization program, that are docketed after March 14, 1978. All aspects of the position, with the exception of reasonable and justified deviations from IEEE 279 requirements, will be applicable to CP, OL, ML, PDA, and FDA applications docketed prior to March 14, 1978 but for which the licensing action has not been completed as of March 14, 1978. Holders of appropriate PDA's will be informed by letter that all aspects of the position with the exception of IEEE 279 will be applicable to their approved standard designs and that such designs should be modified, as necessary, to conform to the position. Staff approval of proposed modifications can be applied for either by application by the PDA-holder on the PDA-docket or by each CP applicant referencing the standard design on its docket.

The following guidelines may be used, if necessary, to alleviate impacts on licensing schedules for plants involved in licensing proceedings nearing completion on March 14, 1978:

1. Those applicants issued an OL during the period between March 14, 1978 and a date 12 months thereafter may merely commit to meeting the position prior to OL issuance but shall, by license condition, be required to install all required staff-approved modifications prior to plant startup following the first scheduled refueling outage.

2. Those applicants issued an OL beyond March 14, 1979 shall install all required staff-approved modifications prior to initial plant startup.

3. Those applicants issued a CP, PDA, or ML during the period between March 14, 1978 and a date 6 months thereafter may merely commit to meeting the position but shall, by license condition, be required to amend the application, within 6 months of the date of issuance of the CP, PDA, or ML, to include a description of the proposed modifications and the bases for their design, and a request for staff approval.

4. Those applicants issued a CP, PDA, or ML after September 14, 1978 shall have staff approval of proposed modifications prior to issuance of the CP, PDA, or ML.

D. References

1. NUREG-0138, Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR, to NRR Staff.

CATEGORY 4 MATTERS

A. Regulatory Guides not categorized

Issue Date	Number	Revision	Title
4/74	1.12	1	Instrumentation for Earthquakes
12/75	1.13	1	Spent Fuel Storage Facility Design Basis
8/75	1.14	1	Reactor Coolant Pump Flywheel Integrity
1/75	1.75	1	Physical Independence of Electric Systems
4/74	1.76	0	Design Basis Tornado for Nuclear Power Plants
9/75	1.79	1	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors
6/74	1.80	0	Preoperational Testing of Instrument Air Systems
6/74	1.82	0	Sumps for Emergency Core Cooling and Containment Spray Systems
7/75	1.83	1	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes
11/74	1.89	0	Qualification of Class 1E Equipment for Nuclear Power Plants
12/74	1.93	0	Availability of Electric Power Sources
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 Sub-system 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

DESCRIPTION OF POSITIONS IDENTIFIED AS NRR CATEGORY 4 MATTERS IN ENCLOSURE 4, PARAGRAPH C

Numbering scheme corresponds to that used in Item C of Enclosure 4.

C.1 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

$\mu = 1.3$

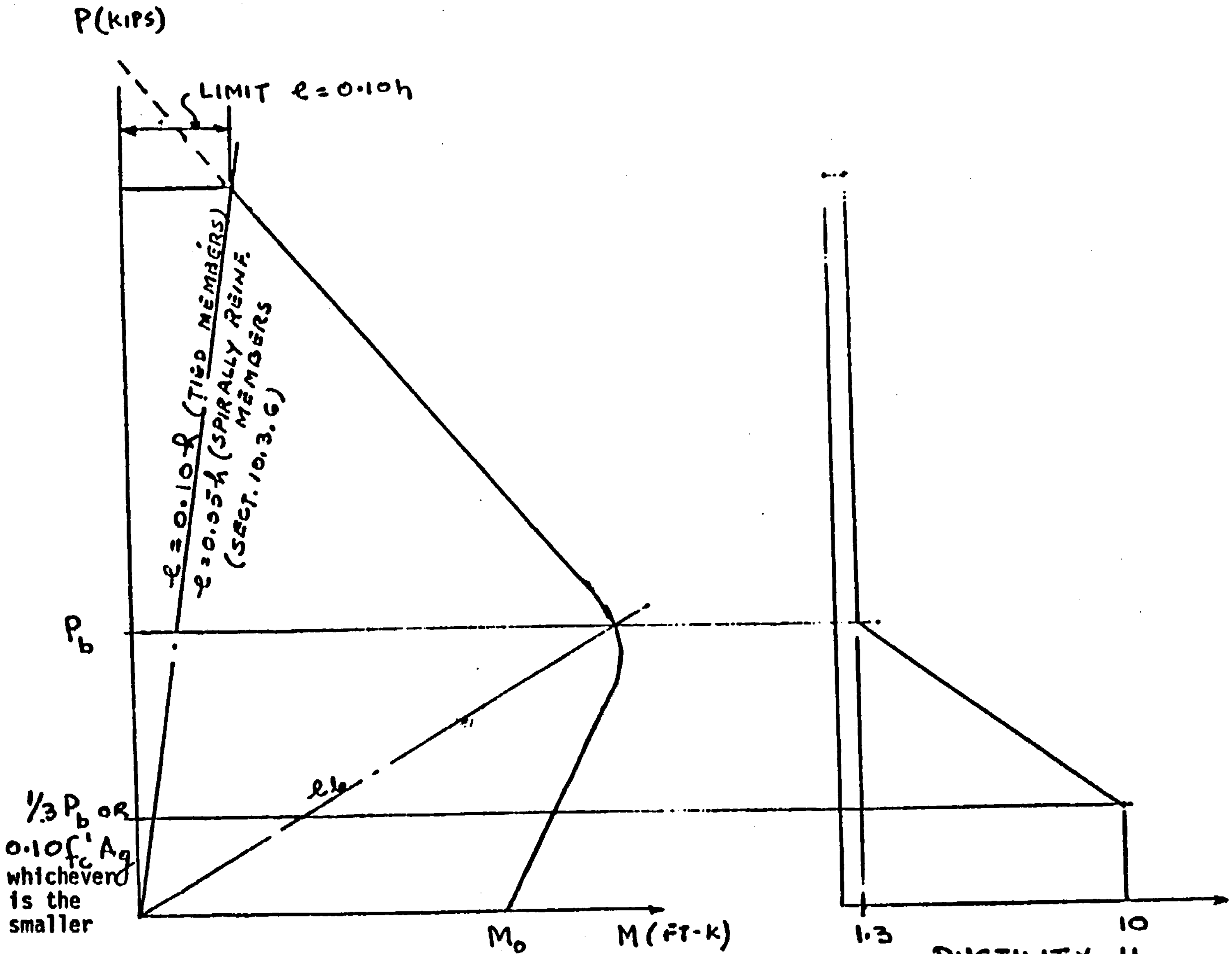

FIG 1. PROPOSED DUCTILITY RATIO FOR BEAM COLUMNS

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

C.2 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 1.60.

C.3 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.

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.

C.4 <u>AIR BLAST LOADS</u> (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.

C.5 TORNADO MISSILE PROTECTION

As an interim measure, the minimum concrete wall and roof thickness for tornado missile protection 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.

C.6 PASSIVE ECCS FAILURES DURING LONG-TERM COOLING FOLLOWING A LOCA

Passive failures in the ECCS, having leak rates equal to or less than those from the sudden 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.

C.7 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.

C.8 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.

C.9 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.

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

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.

C.11 CORE THERMAL-HYDRAULIC ANALYSES

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.

C.12 DEGRADED GRID VOLTAGE CONDITIONS

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.

C.13 ASYMMETRIC LOADS ON COMPONENTS 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.

C.14 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.

C.15 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.

C.16 ENVIRONMENTAL EFFECT OF PIPE FAILURES

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.

C.17 DESIGN REQUIREMENTS FOR COOLING WATER 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.

C.18 <u>WATER HAMMER IN STEAM GENERATORS WITH TOP FEEDRING DESIGN</u>

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.

C.19 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.