Department of Energy
Washington, D.C. 20545
Docket No. 50-537
HQ:S:82:046

JUN 14 1982

Mr. Paul S. Check, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Check:

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION

Reference: Letter, P. S. Check to J. R. Longenecker, "CRBRP Request for Additional Information," dated February 19 and 26; March 11, 15, 23, and 25; and April 9, 1982

This letter formally responds to your request for additional information contained in the reference letter.

Enclosed are responses to Questions CS 210.4, 8, and 10; CS 220.3, 5, 10, 15, 25, 30, and 35; CS 250.1 and 3; CS 410.2, 3, and 19; CS 421.1, 5, 8, 9, 14, 17, 18, 19, 20, and 23; CS 491.18; and CS 490.11 and 35; which will also be incorporated into the PSAR Amendment 69; scheduled for submittal later in June.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of the
Clinch River Breeder Reactor
Plant Project
Office of Nuclear Energy

Enclosures

cc: Service List
Standard Distribution
Licensing Distribution

D001

Question CS210.4

The ASME Section III Code does not require a fatigue evaluation for Class 2/3 piping design. However, for operating at elevated temperature, creep fatigue effects may become severe in Class 2/3 piping. Describe the method used for creep fatigue evaluation for Class 2/3 piping, or justify why such consideration is not needed. Furthermore, verify whether rules in Code Case N-253 are met.

Response

The considerations given to creep-fatigue effects in elevated temperature Class 2/3 piping can be clarified by discussion of the structural criteria available to evaluate these designs. In some elevated-temperature Class 2/3 piping lines, Code Case N-253 is planned to be adopted as an acceptable criteria for demonstrating protection against unacceptable creep-fatigue damage. The project has developed a modification to RDT Standard F9-4T for application to ASME Code Section III, Class 2 and 3 components and piping (Attachment 1). These project-developed criteria, (termed "F9-4TMOD" in the remainder of this response) is the generally applied approach for evaluating creep-fatigue. Both of these approaches will be discussed separately following an identification of where the two approaches are planned to be used. Where possible, an estimate will be provided of the potential for a design satisfying the F9-4TMOD rules to also satisfy Code Case N-253.

Before proceeding with discussions of the two approaches, the first sentence of the NRC question needs to be put into perspective. It is acknowledge that ASME Section III design rules for low temperature Class 2 and 3 piping (NC/ND-3600) do not require an explicit, cycle-by-cycle fatigue evaluation. However, satisfaction of stress range limits for thermal expansion loads (Equation 10) and combined pressure, weight, and thermal expansion stress limits (Equation 11) uses the allowable expansion stress range value, S_A. The determination of S_A is dependent on the cyclic service conditions by introduction of a stress range reduction value, f. The value of f was developed using fatigue test data and accounts for all temperature cycles during life. Since temperature cycling is the principle cyclic load for low-temperature piping, it is concluded fatigue is addressed in the low-temperature Class 2 and 3 piping rules. The only shortcoming of the low-temperature rule approach might be the absence of "thermal gradient" stress in the stress range evaluations. As will be seen, these gradient stresses are picked up in most of the elevated temperature evaluations.

For organizational clarity, four operating regimes for the Class 2/3 piping are defined as follows:

Regime L Piping - Piping that will never experience elevated temperatures under Normal, Upset, or Emergency operating conditions, i.e., it's entire life, except possibly for Faulted events, is spent at temperatures for which stress values exist in ASME B&PV Code, Section III, Appendix I.

Regimes E1, E2, E3 Piping - Piping that will at some time during Normal, Upset, or Emergency conditions experience elevated temperatures.

<u>Regime E1 Piping</u> - Piping that will not be subjected to significant cyclic duty, i.e., it can meet certain exemption criteria from cyclic damage evaluations.

<u>Regime E2 Piping</u> - Piping that will not be subjected to significant elevated temperature service, i.e., it can meet certain material time-temperature limits that preclude significant creep effects from occurring.

<u>Regime E3 Piping</u> - Piping that will be subjected to significant elevated temperature service.

The majority of the Class 2/3 piping falls into Regime L. The low-temperature (NC/ND-3600) cyclic evaluation methods have been discussed, they address fatigue and will not be discussed further since the NRC question is basically concerned with Regimes E1, E2, and E3.

A list of elevated temperature Class 2/3 lines are:

1) The Steam Generator System (SGS) water/steam piping which is a Class 3 piping system is anticipated to be in Regime E1 and thus exempted form creep-fatigue evaluations. The exemption criteria contained in the piping specification are identical to those contained in Appendix A of Code Case N-253. Should it prove necessary to perform a creep-fatigue evaluation, the criteria of Code Case N-253 would be considered for implementation.

2) The Intermediate Heat Transport System (IHTS) hot leg and super heater dump lines downstream of the second valves are Class 3 piping lines which are expected to be in Regime E2. They have been optionally upgraded to Class 2. There exists one emergency design event with fluid temperatures in excess of 800°F. It is expected this event will be exempt from elevated temperature creep and fatigue evaluation. If elevated temperature creep and fatigue evaluations become necessary, they will be performed in accordance with F9-4TMOD.

3) The SGS leak detection module piping between the sodium isolation valve and the LDS modules is instrument piping optionally upgraded to Class 3 and expected to be in Regime E2. Design requirements are not finalized yet but it is anticipated the Code Case N-253 rules will be invoked.

4) The Steam Generator Auxiliary Heat Removal System (SGAHRS) piping between the superheater and the vent control valve is Class 3 piping expected to be in Regime E3. It will be designed to Code Case N-253.

5) The SGS leak detection module piping between the IHTS and sodium isolation valve is Class 2 piping which has been optionally upgraded to Class 1 and will be designed to Code Case 1592 creep-fatigue criteria.

6) The IHTS main and auxiliary piping upstream from the second valve is Class 2 and 3 piping which has been optionally upgraded to Class 1 and will be designed to Code case 1592 creep-fatigue criteria.

7) The Impurity Monitoring and Analysis system (IMAS) and Inert Gas Receiving and Processing System (IGRPS) contain several lines in Regime E2 which will be designed to F9-4TMOD. These lines include:

   a) Primary sodium sampling loop
   b) Sodium sampling line to and from the Intermediate Sodium Characterization System
   c) Equalization line between the primary sodium overflow vessel and the reactor vessel
   d) Primary pump equalization lines to the reactor equalization line
   e) Equalization and overflow vent lines to the vapor condensor
   f) Primary pressure relief line from the equalizer line to the overflow vessel
   g) The overflow vessel vent line.

8) The entire Intermediate Sodium Processing System (ISPS) except for dump lines is in Regime E2 and is being designed to F9-4TMOD.

9) The IGRPS inlet lines to the vapor condensors, drain lines from the vapor condensors, and drain lines from the Intermediate Heat Exchanger are in Regime E3 and are being designed to F9-4TMOD.

10) The Auxiliary Liquid Metal System (ALMS) discharge piping from the primary sodium cold traps in the overflow makeup circuit will be designed per the Regime E3 requirements of F9-4TMOD. Additionally, this piping will have insignificant thermal transient stresses and is in Regime E2, per the time-temperature limits of both F9-4TMOD and Code Case N-253, thus the criteria provides essentially the same protection as the Regime E2 coverage of Code Case N-253.

11) Low-stressed Type 304SS portion of the ALMS inlet piping to the primary sodium cold traps which are in Regime E3 will be designed to the Code Case 1592 elastic method creep-fatigue criteria. High-stressed portions utilize Type 316SS and are evaluated per the Regime E3 per the time-temperature limits of both F9-4TMOD and Code Case N-253, which with the insignificant thermal transient stresses in this line, provides essentially the same protection as the Regime E2 coverage of Code Case N-253.

Discussions will now center of the various evaluative methods for creep-fatigue used in the Regime E1, E2, and E3 piping. Discussions will not be provided of Code Case 1592 methods since the NRC question is concerned with Class 2/3 methods and Code Case 1592 is a Class 1 methodology, which specifically addresses creep-fatigue.

<u>Regime E1 (Insignificant Cyclic Service) Evaluative Method</u>

The Code Case N-253 Appendix A creep-fatigue exemption criteria can be summarized as follows:

> Creep-fatigue evaluations are not required when the maximum peak strain range during service is below the permitted value for 10 cycles from the Code Case 1592 continuous cycling fatigue curve <u>and</u>, less than 25 "significant" load cycles exist during Level A or B service. A detailed

definition of "significant" is provided to assure consistency in rule application. Although Code Case N-253 limits this exemption to -3200 vessels, a similar 25-cycle exemption exists in F9-4TMOD. A justification for the F9-4TMOD 25-cycle exemption criteria for any component is also provided in F9-4TMOD and with only minor modification can be used to justify use of the Code Case N-253 exemption for piping.

Regime E2 (Insignificant Creep Effects) Evaluative Methods

RDT F9-4TMOD

The Class 1 elevated temperature Code rules (Code Case 1592) recognize under certain combinations of stress level, time, and temperature, that creep effects are minimal, and these effects on cyclic life can be addressed by simple modifications to the low-temperature design methodology. The Class 1 time-temperature limit (T-1325) which defines this regime was adopted by the CRBR Project as Appendix E to F9-4TMOD. Satisfaction of this time-temperature limit allows the piping designed to modify and utilize the low-temperature Class 1 design methodology (NB-3600), which includes an explicit, cycle-by-cycle fatigue evaluation (Equation 11) plus consideration of thermal gradient stresses, not normally included in Class 2/3 evaluations. The modification also includes use of the Code Case 1592 "hold-time" fatigue curves (Figure T-1430) which has built into them the worst-case effects of creep on cyclic damage. Therefore, potential creep-fatigue is directly accounted for in this F9-4TMOD design route, even though creep effects are restricted to minimal levels.

Code Case N-253

The Code Case N-253 design route includes a time-temperature limit similar to F9-4TMOD and Code Case 1592; however, if met, it permits a modified Class 2/3 (NC/ND-3600) methodology to be used. Modifications do not require thermal gradient stresses to be considered nor explicit, case-by-case fatigue evaluations to be performed. The logic for this approach is that the time-temperature exclusion criteria means that creep effects are not significant and that piping in this category is really an extension of Regime L.

Regime E3 (Potentially Significant Creep Effects) Evaluative Methods

RDT F9-4TMOD

Two evaluation approaches are permitted in F9-4TMOD:

1) Meet the Code Case 1592 creep-fatigue criteria either using elastic analysis methods (requiring elastic analysis method strain limits also be met) or using inelastic analysis methods (RDT F9-5T). This directly addresses creep-fatigue damage in a manner identical to that of a Class 1 component without upgrading the design to Class 1.

2) Perform a low-temperature Class 2 (NC-3600) evaluation using Section VIII Division I allowables (per Code Case 1481), but reduce the S_A value of NC-3611.2 by the largest value of

$$\frac{1}{2} \frac{E\alpha}{2(1-\nu)} \Delta T_1 + C_3 E_{ab} (\alpha_a T_a - \alpha_b T_b)$$

which occurs at each location during the specified Normal, Upset, and Emergency operating conditions. This approach is thus similar to that used by B31.1 at elevated temperature, but it also includes an accounting of the potential deleterious effects of thermal gradients in addition to restrained thermal expansion.

Code Case N-253 provides the following approach. The design rules for low-temperature piping (NC/ND-3600) must be met with the following modifications:

1) The stress reduction factor, f, is determined using Appendix B to Code Case N-253, where fatigue factors are determined as a function of material and temperature.

2) After excluding 25 cycles from evaluation, a limit is placed on acceptable thermal expansion plus thermal gradient stress range.

3) A modification is made to the NC-3600 Equation 11 allowable on thermal expansion range (to be the lessor of the current limit and a new limit based on material yield strength).

Satisfaction of Code Case N-253 Rules

As noted previously some portions of the CRBR Class 2/3 piping are or will be evaluated directly in accordance with the rules of Code Case N-253. Other portions of the Class 2/3 piping are or will be evaluated using RDT F9-4TMOD or optionally upgraded to Class 1.

In summary, a methodology has been developed and described in the above in order to provide creep-fatigue evaluation for ASME Class 2/3 piping. This methodology is structured to the specific piping in question. This pre-empts the question of across the board compliance to Code Case N-253.

Attachment 1 Q CS 210.4

A CRBRP MODIFICATION TO RDT STANDARD F9-4
FOR APPLICATION TO ASME CODE SECTION III,
CLASS 2 and 3 COMPONENTS AND PIPING

May 1975

0. FORWARD

In the title, and in subsections 0.1, 0.1.1, 0.2 and 0.5.4, change the phrase

"Code Cases 1592, 1593, 1594, 1595 and 1596"

to read,

"Code Cases 1481, 1592, 1593, 1594, 1595, and 1596"

In 0.1 Scope, change to read, ". . . Classes 1, 2 and 3 nuclear components. . ".

In Subsection 0.5.2, change to read: Code Case 1592-2

Code Case 1595-1

In Subsection 0.5.2, add the following:

"Code Case 1481	Elevated Temperature Design of Section III Class 2 and 3 Components".
"Code Case 1606-1	Stress Criteria, Section III Classes 2, 3 Piping Subject to Upset, Emergency, and Faulted Operating Conditions."
"Code Case 1607-1	Stress Criteria for Section III Class 2 and 3 Vessels Designed to NC/ND-3300 Excluding the NC-3200 Alternate".
"Code Case 1635-1	Stress Criteria for Section III, Classes 2 and 3, Valves subject to Upset, Emergency and Faulted Operating Conditions".

In subsection 0.5.4, change the phrase

"to 'Code Case 1592-0' (or 1593-0, 1594-0, 1595-0, 1596-1)"

to read,

"to 'Code Case 1592-2' (or 1593-0, 1594-0, 1595-1, 1596-1, 1481-0)"

5.0 SUPPLEMENTS TO CODE CASE 1481

For RDT F9-4 applications, any reference in Code Case 1481 to the rules of subsections NC or ND shall mean the rules as supplemented by RDT E15-2 NC and RDT E15-2 ND, respectively. Where this supplement references Code Cases 1592 it means the requirements of Case 1592 as supplemented by the applicable portion of RDT F9-4. Where this supplement references NB-3000 it means the requirements of Article NB-3000 of Section III as supplemented by RDT E15-2B.

The rules of Code Case 1481 shall be supplemented by the following additional requirements.

(2) Delete this paragraph in its entirety.

Add the following additional paragraphs:

(5) Applicability

When this standard is invoked for Class 2 and 3 Components, it is applicable only to vessels, piping, and valves. Electro-magnetic pumps may be treated as either piping or vessels.

These rules apply only to Class 2 and 3 components constructed from Types 304 and 316 austenitic stainless steel, Ni-Cr-Fe Alloy 800H, and 2-1/4 Cr-1 Mo ferritic steel permitted in Code Case 1592. For components constructed from other materials, specific design rules shall be provided in the Design Specification. When the temperature of integral component supports exceeds the upper temperature limit of Section III for the component, the component boundary shall be extended to include the elevated temperature portion of the support. These rules are not applicable to components containing pad type nozzles or non-integral attachments. Socket welds shall not be used for Class 2 or 3 Components which contain sodium or radioactive fluids with the sole exception of instrumentation (or control) lines.

(6) Basic Requirements

The supplemental Class 2 and 3 requirements of this standard shall be applied only to components which meet the requirements of Section III via paragraph (1) of Code Case 1481. For Upset, Emergency, and Faulted Operating Conditions, Code Cases 1607, 1635, and 1606 shall apply for vessels, valves, and piping respectively. The requirements of Code Cases 1607, 1635, and 1606 shall be satisfied using the S values from Section VIII, Division 1, per paragraph (1) of Code Case 1481.

The requirements of Appendix B of this standard shall be satisfied in addition to the external pressure limits of NC/ND-3133. Prior service experience or experimental demonstrations may be used in lieu of analysis. When a buckling evaluation of an identical component for equivalent or more severe service to Code Cases 1331-5, 1331-6, 1331-7, 1331-8, or 1592 is available, and when a supplemental stress report (certified by a Registered Professional Engineer) is prepared which reconciles the previous evaluation with the current specification, the requirements of Appendix B are satisfied. The evaluation to the particular Code Case must be complete and shall demonstrate satisfaction of all the Code Case buckling requirements.

Only the Normal Operating Conditions need be considered when determining the most severe condition of coincident pressure and temperature per NC-3112.1 and ND-3112.1.

The owner shall supply the modifications to the design limits which are appropriate for the materials of construction in the specified service environment.

(7) Supplemental Requirements:

Exemption from Ratchetting and Creep-Fatigue Analysis

Ratchetting and creep-fatigue evaluations of the specified Normal, Upset, and Emergency Operating Conditions are required unless one of the following requirements is satisfied:

a) The number of significant load cycles is less than 25. Appendix A of this standard defines the term "significant". This procedure may be used only where the peak strain range is less than the limit given in Appendix A of this standard.

b) Prior servic of equivalent severity is used as the basis for demonstrating structural integrity. Appendix C of this standard describes the "Prior Serv Report" which shall be prepared when this procedure is utilized.

c) Testing of identical components under equivalent or more severe conditions is utilized as the basis for demonstrating structural integrity. Appendix D of this standard describes the Experimental Test Report which shall be prepared when this procedure is utilized.

d) When an evaluation of an identical component for equivalent or more severe service to Code Cases 1331-5, 1331-6, 1331-7, 1331-8, or 1592 is available, and when a Supplemental Stress Report (certified by a Registered Professional Engineer) is prepared which reconciles the previous evaluation with the current specification, these requirements are satisfied. The evaluation to the particular (1331 or 1592) Code Case must be complete and shall demonstrate satisfaction of all of the Code Case ratchetting and creep-fatigue requirements.

e) For Class 3 components only, the owner may, via an explicit statement in the Design Specification, exempt the component from ratchetting and creep-fatigue evaluation.

(8) Supplemental Requirements:

Ratchetting and Creep-Fatigue Analysis

When ratchetting and creep-fatigue evaluations are not exempted by (7) above, one of the following requirements shall be satisfied:

a) An evaluation which demonstrates compliance with all of the requirements of T-1300 and T-1400 of Code Case 1592 as supplemented by RDT Standard F9-4. The alternate procedures in RDT F9-5 may be used. Note that the use of an elastic creep-fatigue evaluation procedure of T-1430 of Code Case 1592 is permited only when the limits of T-1320 are satisfied. In NT-1325 Test No. 4(a) the extrapolation may be performed using the maximum slope at the current metal temperature.

b) The temperature-time limits of Appendix E of this standard and the appropriate requirement below:

Vessels: All of the requirements of NC-3200 with the elevated temperature S_m values obtained from Appendix I-14 of Code Case 1592 and with the design fatigue curves extended to elevated temperatures via Appendix G of this standard. If NC-3219.2 is utilized, the modification of Appendix F of this standard shall be employed. NC-3219.3 shall not be used.

Valves: The exemptions of Conditions A or B of NC-3219.2 as modified by Appendix F of this standard shall be satisfied, or a detailed fatigue analysis shall be made in accordance with the rules of Appendices XIII and XIV where the S_m values at elevated temperatures shall be obtained from Appendix I-14 of Code Case 1592 and the design fatigue curves at elevated temperatures shall be obtained via Appendix G of standard, or the limits of NB-3512.2 (a) or (c) using, at elevated temperatures, the S values from Division 1, Section VIII per paragraph (1), the pressure-temperature ratings of ANSI B16.5, and the fatigue limits of Appendix G of this standard.

Piping: The limits of NB-3600 using, at elevated temperatures, the S_m values from Code Case 1592 and the design fatigue curves of Appendix G of this standard.

c) For piping, the limits of NC-3600 when the basic allowable stress values at elevated temperatures are obtained from Division 1, Section VIII per paragraph (1) and the S_A value of NC-3611.2 (c) is reduced by the largest value of

$$1/2 \left[\frac{E\alpha}{2(1-\nu)} \mid \Delta T_1 \mid + C_3 E_{ab} \mid \alpha_a T_a - \alpha_b T_b \mid\right]$$

which occurs at each location during the specified Normal, Upset, and Emergency Operating Conditions (the terms are defined in NB-3653.2).

(9) Design Reports

Design reports shall be prepared and submitted to the owner which document all of the evaluations performed in addition to the minimum required for stamping by Subsections NC and ND and Code Case 1481. These reports shall follow the ASME Code guidelines for Stress Reports. The reports shall be certified by a Registered Professional Engineer as meeting the requirements of the Design Specification.

The fabricator shall provide the justification for the fatigue limits employed for numbers of cycles greater than 10^6.

APPENDIX A

Significant Load Cycles

A load cycle is "significant" when any of the following is true:

a) The variation in the primary stress is greater than 25% of the maximum allowable primary stress as defined by Code Case 1481.

b) The secondary stress range is greater than 50% of the limiting value ($2S_y$ at Section III temperatures or T-1300, Code Case 1592 at elevated temperatures).

c) The estimated peak stress (K_T = 2.5 when local structural discontinuities exist, unless otherwise justified) is greater than twice the allowable stress amplitude at 10^6 cycles from Figure I-9 of Section III (after the environmental effect correction is applied to the design fatigue curve.)

This procedure is invalid when the maximum estimated peak strain range from any one cycle exceeds the maximum allowable value for 10 cycles as obtained from the design fatigue curve (Figure T-1430 of Code Case 1592) at the maximum metal temperature of the cycle. When the maximum metal temperature is below the value for which Code Case 1592 provides design fatigue curves, the allowable strain range may be obtained from Figure I-9 of Section III (Subsection NA) by dividing the allowable stress amplitude, S_a, by one half of the Figure I-9 Young's Modulus.

APPENDIX B

Buckling Limits

1.0 General Requirements

Where buckling is a potential failure mode, the following minimum design factors shall be maintained.

1.1 The stability limits in NC-3133 and ND-3133 of Section III pertain only to specific geometrical configurations under specific loading conditions. These Section III limits include the effects of initial geometrical imperfections permitted by fabrication tolerances. However, Section III limits do not consider the effects of creep due to long-term loadings at elevated temperatures and the effects of the other loads or geometries. These rules provide additional limits which are applicable to general configurations and loading conditions that may cause buckling or instability due to time-independent creep behavior of the material. These additional limits are applicable to all specified Design and Operating conditions.

1.2 For the limits specified here, distinction is made between load-controlled buckling and strain-controlled buckling. Load-controlled buckling is characterized by continued application of an applied load in the post-buckling regime leading to catastrophic failure as exemplified by collapse of a tube under external pressure. Strain-controlled buckling is characterized by the immediate reduction of strain-induced load upon initiation of buckling, and by the self-limiting nature of the resulting deformations. Even though it is self-limiting, strain-controlled buckling must be avoided to guard against failure by fatigue, excessive strain, loss of functionality due to excessive deformation, interaction with load-controlled instability.

1.3 For conditions under which strain-controlled and load-controlled buckling may interact, the Load Factors applicable to load-controlled buckling shall be used for the combination of load-controlled and strain-controlled loads to guard against buckling in the interactive mode.

1.4 For conditions where significant elastic followup may occur, the Load Factors applicable to load-controlled buckling shall also be used for strain-controlled buckling.

1.5 For load controlled buckling the effects of initial geometrical imperfections and tolerances shall be considered in the time-independent calculations of paragraph 2.1 of this appendix; and the effects of geometrical imperfections and tolerances, whether initially present or induced by service, shall be considered in the time-dependent calculations of paragraph 2.2 of this appendix.

1.6 For purely strain-controlled buckling the effects of geometrical imperfections and tolerances, whether initially present or induced by service, need not be considered in calculation of the instability strain. However, if significant geometrical imperfections are present initially, enhancement due to creep may cause excessive deformation or strain. These effects shall be considered in the application of deformation and strain limits.

1.7 The expected minimum stress-strain curve for the material at the specified temperatures shall be used. The expected minimum value may be obtained by normalizing the appropriate average hot tensile curve of Fig. T-1800 of Code Case 1592 to the minimum yield strength given in Fig. I-14.5 of Code Case 1592.

1.8 The limits of both 2.1 and 2.2 shall be satisfied for the specified Design and Operating conditions.

2.0 Buckling Limits

2.1 Time-Independent Buckling

For load-controlled buckling, the Load Factor, and for strain-controlled buckling, the Strain Factor; shall equal or exceed the values given in Table B-1 for the specified Design and Operating conditions to protect against time-independent (instantaneous) buckling.

TABLE B-1 TIME-INDEPENDENT BUCKLING LIMITS

	Load Factor(1)	Strain Factor(1) (3)
Design Conditions	3.0(2)	1.67
Operating Conditions		
Normal	3.0	1.67
Upset	3.0	1.67
Emergency	2.5	1.4
Faulted	1.5	1.1
Testing(4)	2.25	1.67

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(1) Load (Strain) Factor. = Load (strain) which would cause instant instability at the design or actual operating temperature. ÷ Design or expected load (strain).

(2) Changes in configuration induced by service need not be considered in calculating the buckling load.

(3) For thermally induced strain-controlled buckling, the Strain Factor is applied to loads induced by thermal strain. To determine the buckling strain, it may be necessary to artificially induce high strains concurrent with the use of realistic stiffness properties. The use of an "adjusted" thermal expansion coefficient is one technique for enhancing the applied strains without affecting the associated stiffness characteristics.

(4) These factors apply to hydrostatic, pneumatic, and leak tests. Other types of tests shall be classified according to-3113.7 of Code Case 1592.

2.2 Time-Dependent Buckling

To protect against load-controlled time-dependent creep buckling, it shall be demonstrated that instability will not occur during the specified lifetime for a load history obtained by multiplying the specified Operating condition loads by the factors given in Table B-2. A design factor is not required for purely strain-controlled buckling because strain-controlled loads are reduced concurrently with resistence of the structure to buckling when creep is significant. The time-temperature limits of Appendix E of this standard may be used to determine whether time-dependent buckling need be considered.

TABLE B-2 TIME-DEPENDENT LOAD-CONTROLLED BUCKLING FACTORS

Operating Conditions	
Normal	1.5
Upset	1.5
Emergency	1.5
Faulted	1.25

APPENDIX C

Prior Service Experience

A Prior Service Experience report may be prepared to demonstrate that the prior service experience of identical components was at least as severe as that specified in the Design Specification. In this manner, successful prior experience can be used in lieu of ratchetting and creep-fatigue analysis for cases where the prior service is at least as severe as that currently specified. The Prior Service Experience Report shall provide a comparison of the proposed design and the design with which the successful service experience was accummulated to demonstrate that they are identical. The report shall also demonstrate that the prior service was at least as severe as that currently specified. Among the factors which are to be used to demonstrate equal load severity shall be the maximum metal temperature and the total duration above the temperature limits of Section III. The Prior Service Experience report shall be certified to be completed and correct by a Registered Professional Engineer. The Prior Service Report shall be attached to the Design Report.

APPENDIX D

Experimental Demonstration of Integrity

Experimental tests of components may be used to demonstrate their structural integrity in lieu of a ratchetting and creep-fatigue analysis.

The experimental tests shall be performed using test conditions which are at least as severe as those identified in the Design Specification. The number of tests and degree of test severity (beyond that defined in the Design Specification) shall be sufficient to demonstrate satisfactory component service. An Experimental Test report shall be prepared which relates the test conditions to the specified service conditions and demonstrates that the test results are a good and proper simulation of the specified service conditions. The report shall contain thermo-hydraulic calculations, where applicable, as well as structural calculations. The Experimental Test Report shall be certified by a Registered Professional Engineer to be correct and complete.

APPENDIX E

Temperature-Time Limits

1.0 When these limits are satisfied, creep effects do not need to be accounted for in the evaluation of the primary plus secondary and primary plus secondary plus peak stress intensities. Their use, however, requires that the fatigue limits be modified. Faulted Operating conditions (when specified) need not be considered in these evaluations.

The fatigue damage sum, when used, shall be limited to 0.9 instead of 1.0.

2.0

This limit is satisfied when both

$$\sum_i \left(\frac{t_i}{t_{id}}\right) \leq 0.1 \quad \text{and} \quad \sum_i (\varepsilon_i) \leq 0.2\%$$

where t_i is the total duration of the metal temperature, T_i, during the specified design lifetime and t_{id} is the minimum time to rupture at a stress level of 1.5 times the minimum yield strength at that temperature (1.5 S_y^{min}). The minimum stress to rupture charts and figures of Code Case 1592 are a valid data source. If extrapolation to greater load durations is necessary, the extrapolation shall be performed using the greatest slope for that material on the plot of the logarithm of the minimum stress to rupture vs the logarithm of the time to rupture for that metal temperature. The $\sum_i t_i$ shall equal the specified design lifetime.

The ε_i value is the thermal creep strain which is accumulated due to the imposition of a uniaxial stress of 1.25 times the minimum yield strength (1.25 S_y^{min}) at the associated metal temperature, T_i, during the time duration, t_i. Creep hardening from one time period (t_i) shall not be accounted for in other time periods. The isochronous stress-strain curves of Appendix T of Code Case 1592 may be used to evaluate this limit.

3.0 Figures E-1 through E-4 may be used to evaluate these limits. The specific curves are based on Code Case 1592 S_y^{min} and S_R values plus the thermal creep equations of the Nuclear Systems Materials Handbook.

APPENDIX F

Modifications to NC-3219.2 Fatigue Exemption Rules

When the procedure of NC-3219.2 is employed, the following modifications shall be incorporated.

Condition A

1) The S_m values for elevated temperatures shall be obtained from Code Case 1592.

2) In NC-3219.2(c), the absolute value of ΔT_1 shall be added to the change in metal temperature between two adjacent points, ΔT, before determining the cycle factor.

   The term ΔT_1 is the maximum equivalent linear through the wall temperature difference (see NB-3653.2(b)). All specified Normal, Upset and Emergency conditions shall be considered.

Condition B

1) The reduction in the allowable S_a value beyond 10^6 cycles shall be considered.

2) The allowable S_a value shall be reduced, as shown below, to account for the maximum through the thickness thermal gradient during all specified Normal, Upset and Emergency Operating conditions.

$$S_a{}^1 = S_a - 1/2 \left[\frac{E\alpha}{2(1-\nu)} \mid \Delta T_1 \mid\right]$$

APPENDIX G

Elevated Temperature Design Fatigue Curves

When this Appendix is invoked, the design fatigue curves of Figures T-1430 of Code Case 1592 shall be used, where S_a is one half of the product of the total strain range, ϵ_t, and the Young's Modulus, E, at that temperature.

In all cases the design fatigue curve shall be modified by the appropriate environmental effect corrections prior to use.

In all cases the appropriate reduction, if any, in the allowable stress (or strain) amplitude beyond 10^6 cycles shall be employed.

A Basis for the Twenty-five Cycle Exclusion

This note provides an evaluation of the potential for failure of components which meet the twenty-five cycle creep-fatigue exclusion.

The minimum number of cycles of Figures T-1430 of Code Case 1592 is ten. Assume that twenty-five cycles at a strain range equal to the strain range of Figure T-1430 at ten cycles are specified. The mean failure fatigue curve is at least a factor of twenty on life above the design curve. Thus a strain range which is associated with ten cycles on the design fatigue curve is also associated with (10 x 20) 200 cycles on the mean failure fatigue curve. It is judged that the minimum failure fatigue curve lies not more than a factor of four on life below the mean failure fatigue curve. Thus, a strain range which is associated with ten cycles on the design fatigue curve is also associated with (10 x 20 ÷ 4) 50 cycles on the mean failure fatigue curve. Thus, twenty-five cycles of that strain range is still a factor of two below the lower bound failure fatigue curve.

The effect of temperature is already accounted for by the temperature effects already built into the Figure T-1430 design fatigue curves. The effects of slow straining rates and effects of hold periods at maximum strain are built into the Figure T-1430 design fatigue curves.

Thus, all of the factors thought to be important in determining fatigue life have been considered. A safety factor of two on life exists between the allowable number of significant cycles and the lower bound failure fatigue curve.

JUSTIFICATION FOR USE OF STRESS RUPTURE EXTRAPOLATION PROCEDURE

Introduction

In Appendix E of the CRBRP Modification to RDT Standard F9-4 (for use with Class 2 and 3 equipment) a time-temperature threshold is provided. When the time-temperature combination is below this limit, it is judged that low temperature fatigue and ratcheting rules may be used. Among the criteria used to define this threshold is the locus of points at which the time duration divided by the "allowable time duration" at 1.5 times the minimum yield strength is equal to 0.1. The "allowable time duration" is defined as the minimum time to rupture. Code Case 1592 provides the minimum time to rupture vs. stress and temperature (S_R^{min}) over a range of stress and temperature values. The yield strength varies relatively slowly with temperature while the minimum stress to rupture value is very temperature sensitive. Thus, at low temperatures (~800°F austenitic steels, ~700°F low alloy steels) 1.5 times the minimum yield strength may be <u>smaller</u> than the minimum stress to cause rupture, even at 1.0×10^5 to 3×10^5 hours. Thus, extrapolation of the stress-rupture curves to time above 3×10^5 is necessary.

Objective

The purpose of this note is to document the reasons for employing a stress-rupture extrapolation technique which is less conservative than that in Code Case 1592.

Austenitic stainless steels, among others, can exhibit a slope discontinuity in their stress-rupture behavior. At long lifetimes (or high stress levels, or high temperatures) the linear relationship between the logarithm of the stress vs. the logarithm of the lifetime may decrease abruptly. In the practical use of stress-rupture data there are many cases where low stress applications are encountered. The designer is faced with the task of determining the damage due to these low stress levels. A conservative method for extrapolating the stress-rupture curves was needed. If the curves were extrapolated linearly based on the maximum slope at the temperature of use, the results might be unconservative if a slope discontinuity existed just beyond the maximum lifetime of the curve.

If, however, the curve was extrapolated at a slope equal to the highest value for all temperatures, there is great assurance that the result is conservative.

The second method, extrapolation at the maximum slope for all temperatures is believed to be conservative for all temperatures up to a few hundred degrees below the maximum for the curve. For types 304 and 316 austenitic stainless steel, the maximum metal temperature for which Code Case 1592-3 provides minimum stress-to-rupture values is 1500°F. Since the LMFBR is not expected to operate with metal temperatures above 1200°F, the use of the 1500°F slope to extrapolate 1200°F stress rupture data is expected to be safe.

In the case of time-temperature limits (below which low-temperature ratchetting and fatigue procedures are valid) the use of the stress-rupture extrapolation using the slope at the maximum temperature of the table is unrealistic. At low stress and temperatures, the stress-rupture slope discontinuity is not expected to occur at all. In 304 S/S the initial stress-time dependency of failure is not even observed until lifetimes of greater than 10^4 hours. The existence of a slope discontinuity nearly is not expected. Even if a slope discontinuity were to occur just beyond the Code Case's time limit (3×10^5 hrs) its slope is not expected to be as high as that for 1500°F.

The use of the maximum temperature slope extrapolation method, when applied to the Appendix E threshold, results in a severe and unwarrented design limitation at low service temperatures, which would have an unnecessary cost impact on the CRBRP Project. The Appendix E threshold is shown below:

Allowable Load Duration, Hours

Temperature °F	304 S/S		316 S/S	
	Max. Temp	This Temp	Max. Temp	This Temp
800°F	200,000	2,000,000	900,000	large
850°F	80,000	80,000	400,000	large
900°F	27,000	27,000	140,000	1,200,000
950°F	4,200	4,200	43,000	68,000

Summary

The stress-rupture extrapolation procedure of the Code Case is inappropriate for this use. The only place where it is more restrictive than current metal temperature linear extrapolation procedure is at low temperatures (800-900°F). The reasoning behind the maximum temperature slope extrapolation procedure is not applicable at low temperatures. The result of using the Code procedure would be an unnecessary and expensive reliance on creep analysis for very low temperature applications.

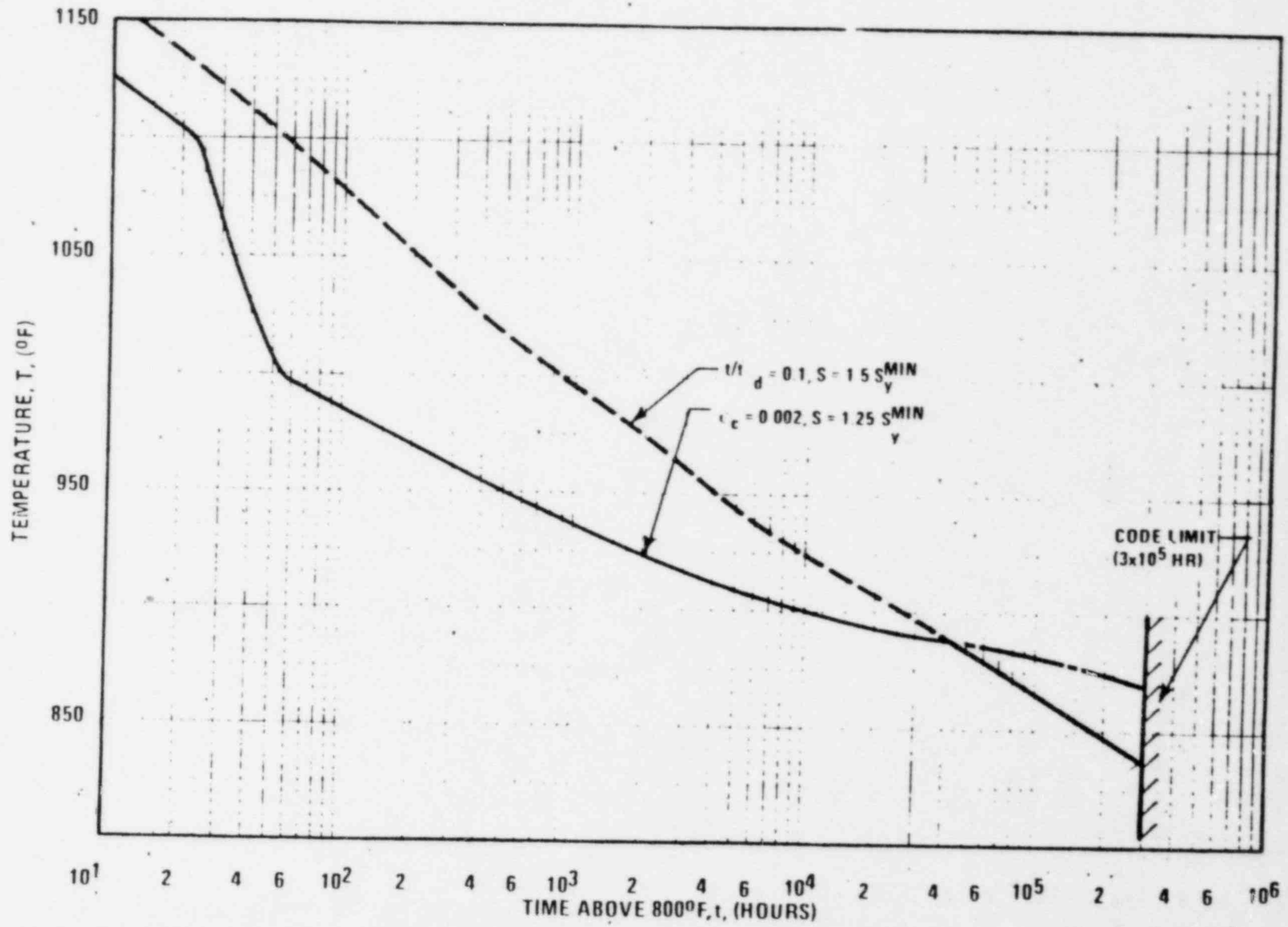

Figure E-1 Temperature Time Limits 304 S/S, Annealed

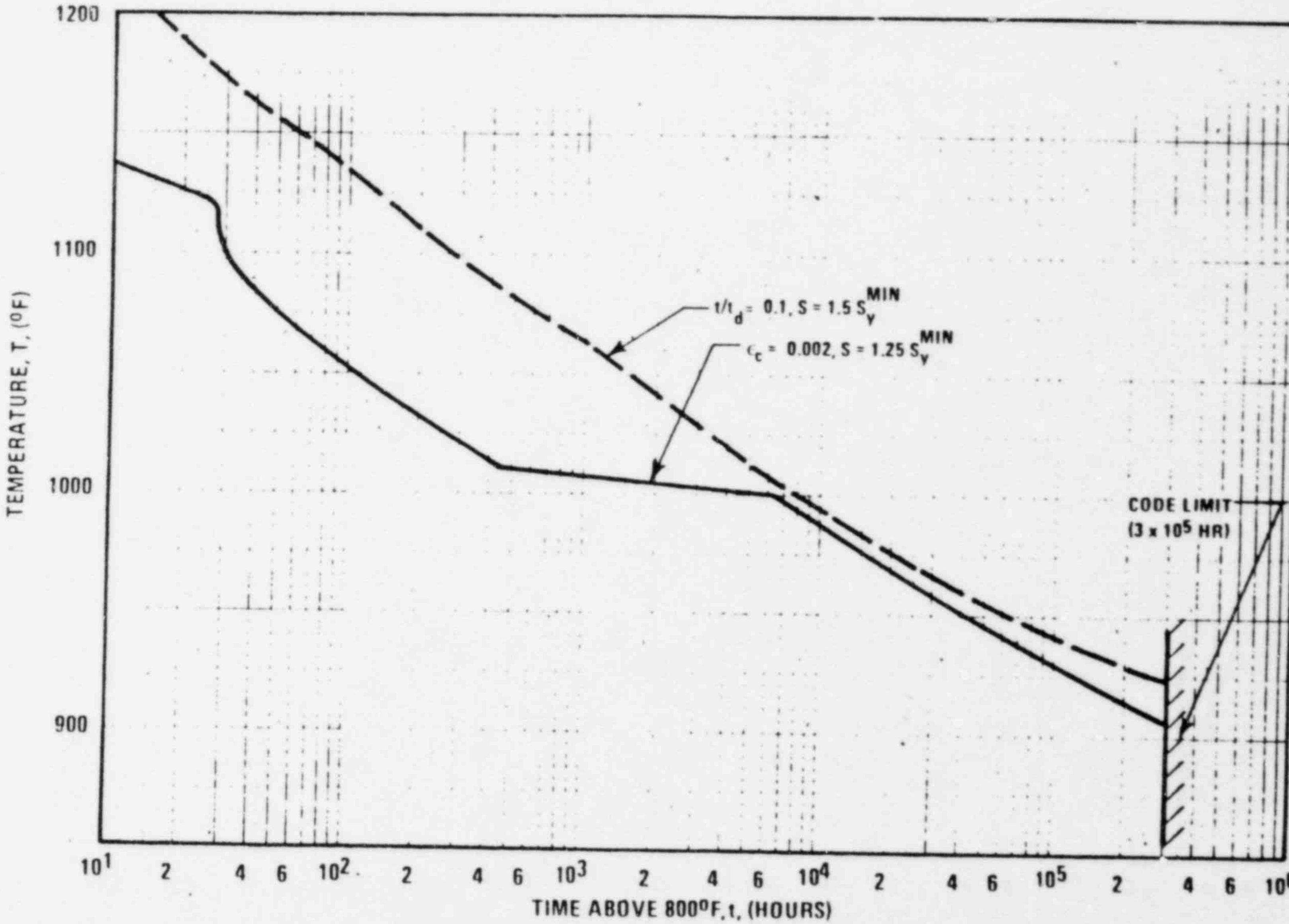

Figure E-2 Temperature - Time Limits 316 S/S. Annealed

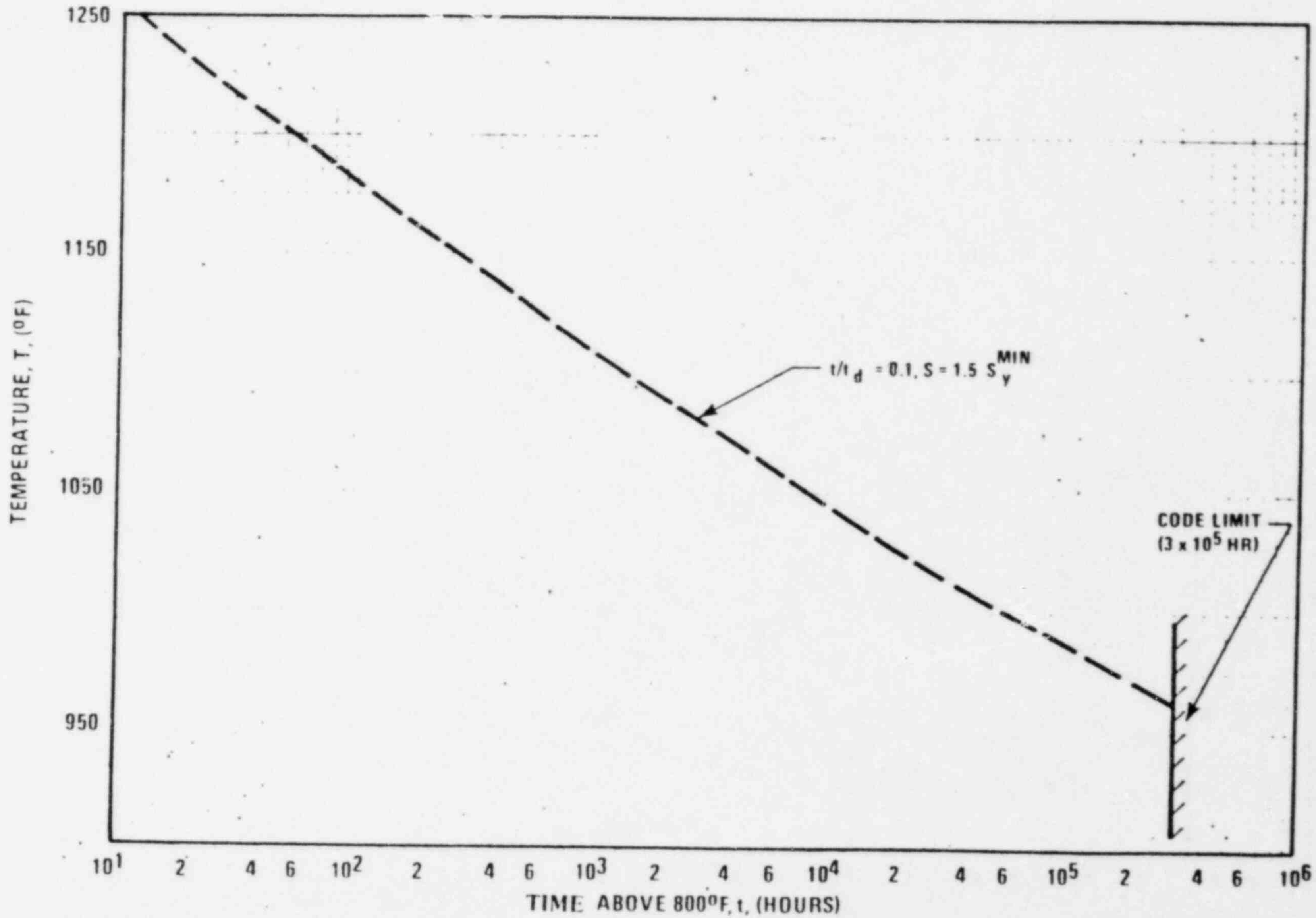

Figure 1-3 Temperature-Time Limits Alloy 800H

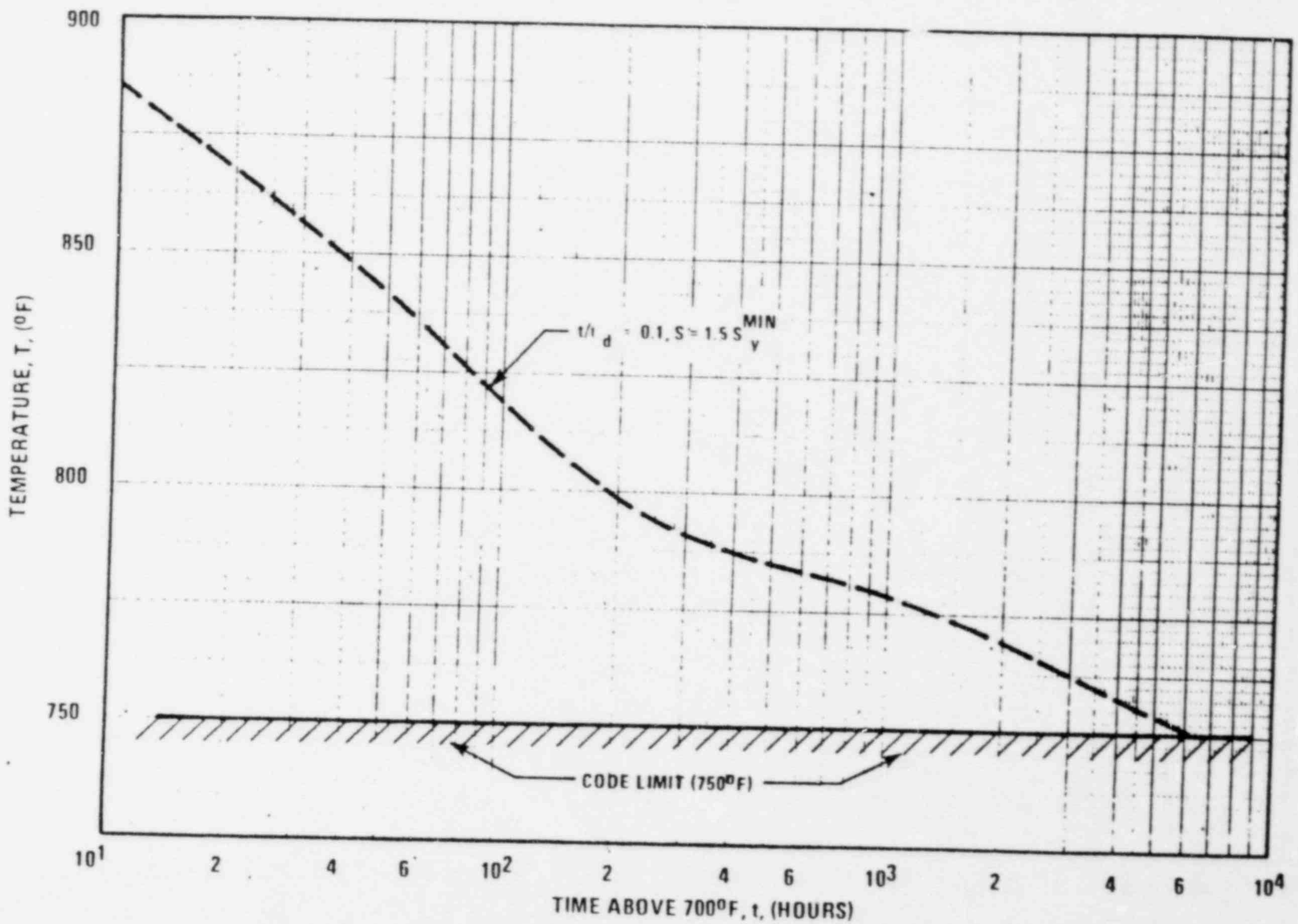

Figure E-4 Temperature-Time Limits 2-1/4 Cr - 1 Mo

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Question OCS 210.8

Provide rules and acceptance criteria used in the design of mechanical component supports, especially those supports for piping at elevated temperature service.

Response

Component supports as defined by ASME Section III, Subsection NF include structural elements, which carry the weight of components or provide them with structural stability, or both. The term includes hangers, supports, braces, snubbers and other devices which are designed to transmit loads from the pressure retaining barrier of the component to the load carrying building structure during any of the specified operating conditions. Therefore, component supports intended to conform to the requirements for Class 1 construction as set forth in ASME Section III, Subsection NB, are governed by Subsection NF, as supplemented by RDT Standards. These documents provide the rules for the materials, design, fabrication, examination and testing of component supports.

Question CS210.10

Describe the basis of "leak without break" criteria used in Section 3.6 of CRBRP-PSAR for the primary and secondary sodium loops. since operating temperature in the hot legs are significantly higher than the cold legs and secondary loops, explain why the same pipe leakage criteria is applicable.

Response

The applicability of the concept of leak before break to the CRBRP hot leg piping in the PHTS cells of the Reactor Containment Building is discussed in the Project's topical report on this subject. This report "Clinch River Breeder Reactor Plant Integrity of Primary and Intermediate Heat Transport System Piping in Containment", WARD-D-0185, was issued in September 1977 and is referenced in Section 3.6 of the CRBRP PSAR. Copies of this report have been recently provided to the CRBRP Program Office at NRC.

Question CS220.3 (3.5.4.5)

On page 3.5-13b ductility ratios for concrete and steel are listed. Some of these ratios are different from those specified in Appendix A to SRP Section 3.5.3. Conformance to SRP Section 3.5.3 ductility ratios is requested unless justification for deviation is provided.

Response:

PSAR page 3.5-13b is revised to incorporate ductility ratios used in the design of concrete and steel structures.

Where:

μ = ductility: ratio defined as the ratio of the deflection of a structure at failure to its deflection at yield.
T = natural period of vibration (seconds)

The Ductility Ratios, which will be used in determining the overall structural response due to missile impact load for various structural components, are given as follows:

DUCTILITY RATIOS (μ)

Reinforced Concrete Members

Maximum value of μ for reinforced concrete members will be as stated in ACI 349, except stringent requirements specified in R.G.1.142 will be used.

Structural Steel Members

a. For tension due to flexure

$\mu_d \leq 10.0$

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

$\mu_d \leq 1.3$

Where l = effective length of the member

r = the least radius of gyration

For columns with slenderness ratio greater than 20

$\mu_d \leq 1.0$

c. For members subjected to tension

$\mu_d \leq 0.5 \frac{e_u}{e_y}$

Where e_u = Ultimate strain

e_y = Yield strain

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Case 2. Missile Not Penetrating the Structural Elements

Williamson and ALvy (Ref. 7) derived a formula for determining a static concentrated load (q_y) equivalent to the force induced by a missile striking a structural element without penetration. the formula is:

$$q_y = \left(\frac{\mu}{2\mu-1}\right) mg \left[1 + \sqrt{1+\left(\frac{2\mu-1}{\mu^2}\right)\left(\frac{2\pi V}{gT}\right)^2} \right]$$

When the last term under the radical and its square root are large when compared to unity, the above equation reduces to

$$q_y = \frac{2\pi mV}{T} \sqrt{\frac{1}{2\mu-1}}$$

Where:

m = Mass of missile (lb-sec.^{2}/ft.)

3.5.5 Missile Barrier Features

Structures housing safety-related systems and components and located above the ground will be designed to resist both the externally and internally generated missiles. The steel Containment structure will be designed with sufficient thickness of steel plate to prevent performation and with stiffeners proportioned to maintain the overall stability. Preliminary evaluations show that a containment vessel shell thickness of 1-1/8" will be capable of accommodating all the potential tornado missiles. Reinforced concrete, Seismic Category I structures, which contain vital equipment will be designed with sufficient strength and ductility to prevent penetration and to absorb impact from the postulated missiles.

Question CS220.5

The major seismic Category I structures of the CRBR plant are supported on a common basemat founded on competent rock with an embedment of 100 ft of back fill. Under such a condition, it appears most appropriate to consider the structures as fixed at the foundation.

The embedment effect can be accounted for by considering the soil-structure interaction between the lateral earth pressure and the structure in contact. The seismic input motion should be applied at the foundation level. The applicant has considered an analysis in which there is soil (rock) structure interaction at foundation level as well as on the lateral side with the seismic input motion applied at the finished grade level. In staff's opinion such an analysis does not represent the realistic condition and the complexity of the analysis as used by the applicant precludes a prior assessment of the adequacy of the method for staff review. As a resolution of staff's concern it is required that seismic Category I structures, systems and components be designed to seismic effects obtained by enveloping the results of applicant's and the fixed base approach as stated above or equivalent.

Response:

The major Category I structures of CRBRP with the exception of the Diesel Generator Building, are supported on a common basemat founded on rock with an average shear wave velocity of 4000 ft/sec. The material above the elevation of the foundation mat (the embedment material) consists of sound and weathered rock, lean concrete fill and compacted Class A backfill.

The input motions were applied at the foundation level and not at grade level (Section 3.7.1.1 of the PSAR).

The justification for using lumped springs and dashpots in lieu of "fixed" base for rock-structure interaction is given below.

1) In the seismic analysis of the CRBRP Nuclear Island, the actual stiffness of the foundation material was evaluated in terms of equivalent springs and dampers and on this basis the seismic analysis was performed.

2) In the CRBRP seismic analysis it was considered that using a fixed based analysis was unwarranted, since due to the large size and stiffness of the structure that is comprised of all the Nuclear Island buildings, some interaction was expected between the foundation rock and the structure. This was confirmed by the calculated responses at the foundation mat which differed from the "free field" responses (Figure Q220.5-1)

   Additional analysis, using a different analytical approach (the Computer Program FLUSH) showed that the free-field and in-structure seismic responses at the foundation level differed, confirming that rock(soil) structure interaction will occur (Figure Q220.5-2).

3) To calculate the foundation springs, because of the irregular layering of the site, and variation of foundation properties, a static finite element method was used. (Section 3.7.1.6 of the PSAR).

4) Calculations using the theoretical half-space equations were also performed (Response to NRC Question 130.53). The results of the elastic half-space calculations verified, within reasonable limits, the values obtained from the static finite element calculations.

5) To account for uncertainity in foundation material properties three sets of spring stiffnesses were calculated: for upper bound, average and lower bound of material properties. Analyses showed that responses with average foundation material properties were enveloped by those for the upper and lower bound. Two complete analyses were then performed using the upper and lower bound material properties and the responses were enveloped.

6) The radiation damping was calculated based on the half-space equations for equivalent material properties deducted from the spring constants obtained by the finite element calculations. This approach was validated by the fact that the results from the finite element and half-space calculations were in good agreement.

SRP 3.7.2 of NUREG-0800 defines various acceptable methods for modeling and analyzing soil structure interaction effects. As described above, the CRBRP seismic design conforms to acceptable methods that are representative of the site geologic conditions and the Nuclear Island structures. Appropriate conservatism has been included in the model to account for a range of foundation material properties that were developed from a rigorous subsurface investigation. The SRP defines "a fixed base assumption" as an acceptable basis for modeling if the structures are supported on rock. Such an assumption is not considered reasonable for CRBRP since analysis has confirmed that interaction will occur and a more realistic and conservative representation of the rock and embedment conditions has been accounted for in the seismic design. The average shear wave velocity of 4000 fps is not characteristic of a hard rock material that would be consistent with a "fixed" base assumption.

The requirement of a fixed base approach for CRBRP is inappropriate and arbitrary. The analysis of a "fixed" base model, therefore, will not be considered for CRBRP.

- o The largest historical earthquake in the tectonic province was assumed to occur in the CRBRP site.
- o SSE maximum ground acceleration was increased from 0.18g to 0.25g.

- Design Response Spectra consist of wide band envelope spectra based on statistical studies of many past earthquake records.

- Artificial acceleration time-histories used in the seismic analysis envelope and for most frequencies are above the Design Response Spectra.

- Floor response spectra are envelopes of two independent analyses using lower and upper bound of soil-rock properties.

- Floor response Spectra were widened at peaks and smoothed.

An independent finite element analysis using the FLUSH program was performed to compare floor response spectra with the CRBRP design spectra. The CRBRP spectra in essance envelopes the calculated spectra and spectra generated from the CRBRP and finite element analyses are very similar (Figure Q220.5-3).

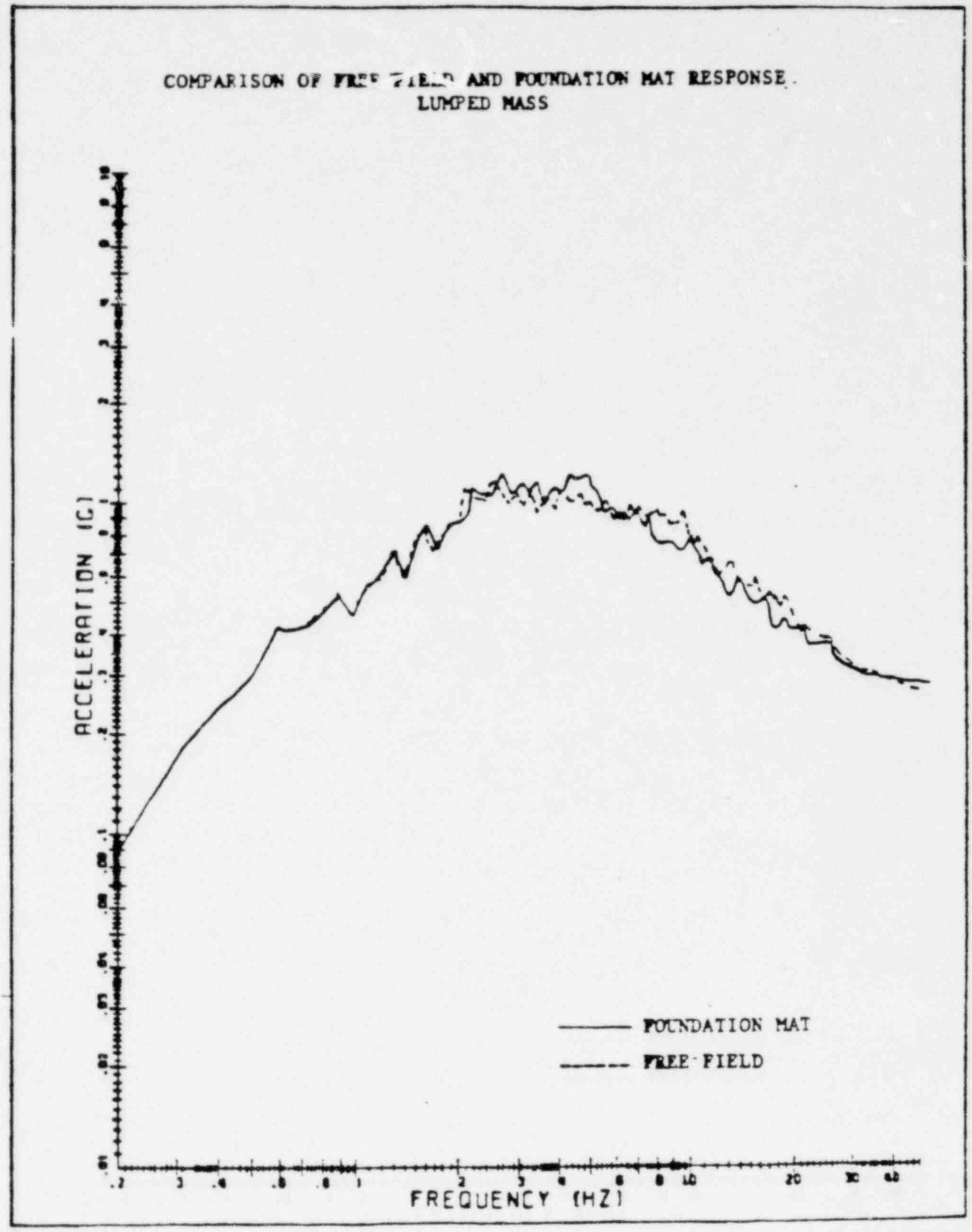

Figure Q 220.5 - 1

Q 220.5-4

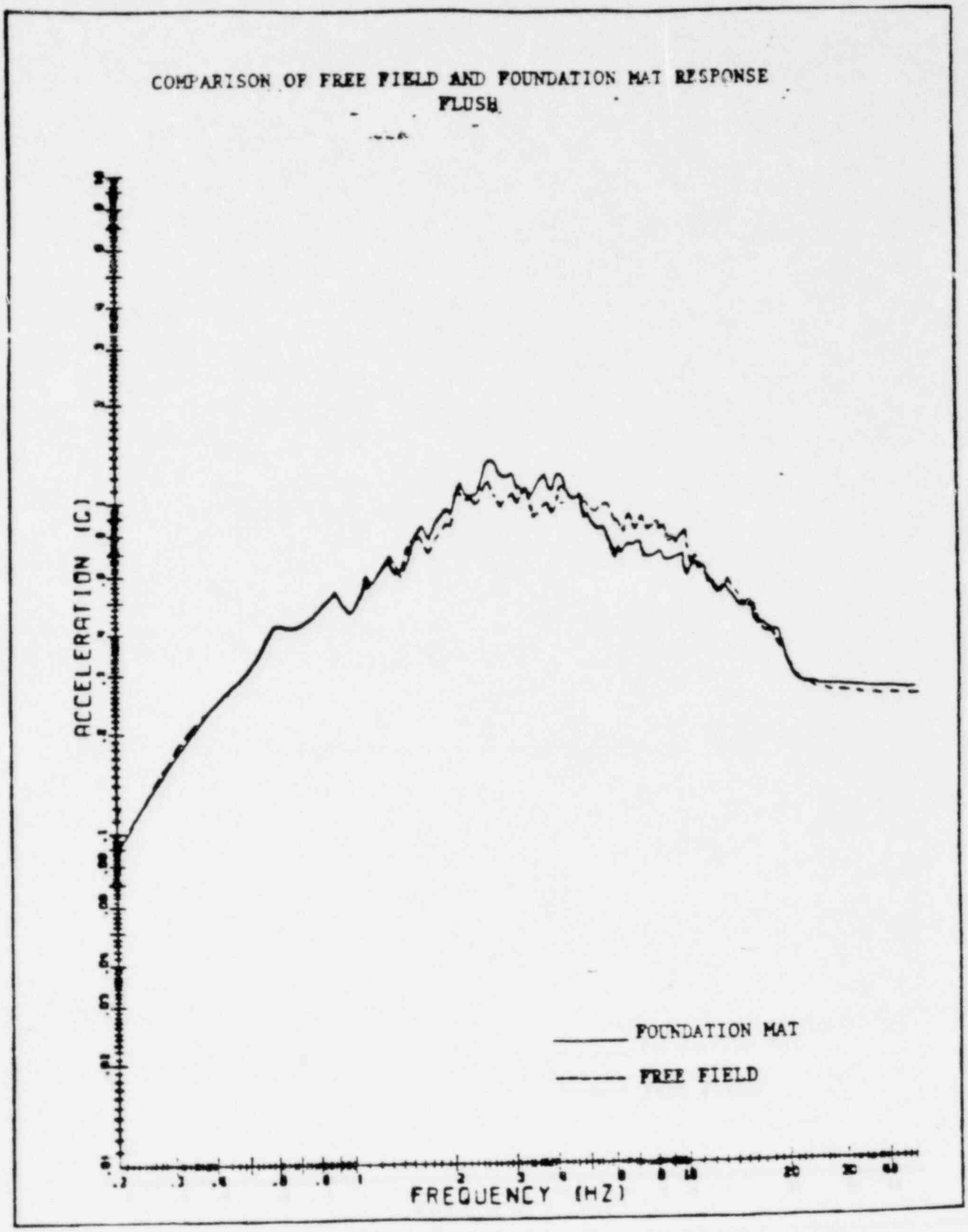

Figure Q 220.5 - 2

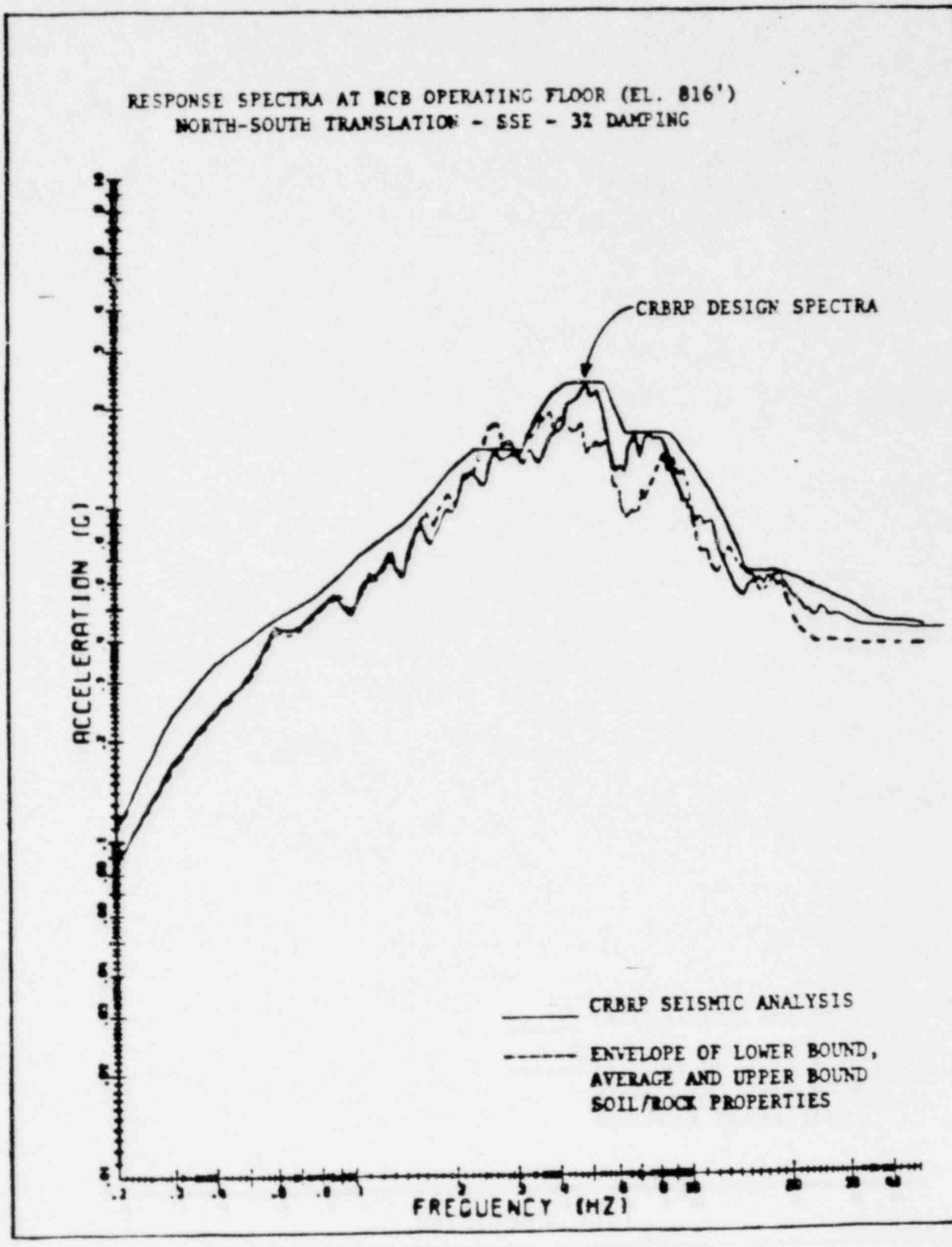

Figure Q 220.5 - 3

Question CS220.10 (3.7.1.6)

In last paragraph on page 3.7-4, you stated that a fixed base approach would be justified. However, in order to account for soil-structure interaction effects, you made a number of simplifying assumptions and also conducted a scoping study to take into account variations in spring constants and damping values. Indicate if you have taken the fixed base condition into consideration in your scoping study since this is more representative of the actual condition.

Response:

See the response to Question CS220.5.

Question CS220.15

On page 3.7-8, it is stated that you perform a non-linear history analysis. Provide more details of such an analysis.

Response

The nonlinear seismic analysis of reactor systems was performed on the Primary Control Rod System (PCRS) since this system was determined to contain significant nonlinearities. The description of this analysis is given in Section 3.7.3.15.3 of the PSAR.

mathematical representation of the system or components. A sufficient number of masses with their appropriate degrees of freedom are used in the model to adequately describe the behavior of the structural system, and to insure an accurate determination of the dynamic response. Significant non-linearities, such as gaps or clearances between PCRS components, are included in the mathematicl model. In this case, a nonlinear time history analysis is performed, which considers the impact forces generated at the gap locations. Non-symmetrical features of geometry, mass, and stiffness, are modeled to include their torsional effects in the analysis. A description of a preliminary reactor system linear model and a preliminary PCRS non-linear model is given in Section 3.7.3.15.

The methods of response spectra analysis and time history analysis are described in a number of publications. A description of these analyses techniques is provided in Appendix 3.7-A.

The system or component is analyzed with the seismic input (floor response spectra or time histories) derived at the particular points of support on the structure. All significant modes of the mathematical model are included in the analysis. The significant, dynamic response modes are those predominant modes which contribute to the total, combines modal response of the system. Other modes, whose inclusion in the square root of the sum of the squares modal summation have anegligible effect on the total response would not necessarily be used. With this procedure the number of modes included will be such that inclusion of additional modes will not result in more than a 10% increase in responses. Where the response spectrum method is used, the individual modal responses are combined by the square root of the sum of the squares, except for closely spaced modes (frequencies less than about 10% apart) where the modal responses are combined by the absolute sum. The analysis is performed independently in each of the two horizontal directions, and the vertical direction. Similar effects obtained for each of the three directions are combined by the square root of the sum of the squares. This is consistent with Regulatory Guide 1.92.

A simplified analysis based on a single mass model or an equivalent static load method may be used when it can be demonstrated that the simplified analysis provides adequate conservatism. For the simplified analysis, the equivalent static force, F_s, is distributed proportional to the mass of the component, and is calculated by the following equation:

$$F_s = 1.5 \, W \, A_s$$

where W is the total weight of the component, and A_s is the maximum peak acceleration of the response spectra, which apply at the points of support of the component. Components whose fundamental frequencies are greater than 33 Hz in any direction, are assumed to be rigid in that direction and may be designed for at least the maximum acceleration at their supports.

All systems and components under the jurisdiction of the ASME Section III Nuclear Power Plant Components Code will be designed to accommodate seismic loadings in combination with other loadings without producing total combined stresses in excess of those allowed by the Code. For elevated temperatures, applicable ASME-III Code Cases and RDT Standards will also apply. Stresses resulting from load combinations which include

Question CS220.25

On page 3.8-1, it is stated that ASME Section III Division 1, 1974 Edition with Addenda through winter 1974 and ASME Section III Division 2, 1975 Edition will be used for the design of the steel containment and the steel lined concrete containment foundation mat respectively. Indicate what will be the effect on the design of the latest editions of the ASME Section III Division 1 and 2 including Code Case N-284 (1980) are used.

Response

The PSAR design was performed to the requirements of the 1974 Code edition specified in the design specifications. The specific criteria related to buckling are described in the PSAR Appendix 3.8-A. The intent of these criteria is similar to the Code Case N-284 criteria, in that these address buckling modes, provide capacity reduction factors and factors of safety, and similar interaction equations for buckling. A significant reanalysis would be required to demonstrate that the Containment Vessel meets the requirements of the new Code and Code Case N-284, however, the applicant has compared the PSAR to the 1980 ASME Code, and has evaluated the significance of the changes. Several of these changes are considered to be of sufficient significance to require additional study. This comparison will be provided by July 15, 1982, and the additional study of the significant changes will be provided by August 30, 1982.

The applicant believes that the intent of N-284 and the 1980 ASME Code has been implemented by the PSAR and the PSAR Appendix 3.8-A, the vessel design is adequate and safe, and that no analysis to the 1980 Code or to Code Case N-284 is necessary.

Question CS 220.30

The ultimate capacity of the steel containment should be addressed.

Response

The ultimate capacity of the steel containment is addressed in the document "CRBRP-3, Hypothetical Core Distruptive Accident Considerations in CRBRP, Volume 2, Assessment of Thermal Margin Beyond the Design Base." A tabulation of the allowable pressures for different temperatures of the material is given in Table 3-10 of the above document. These pressures were calculated with primary membrane stress limits for Service Limit Level D given in NUREG-0800 (SRP 3.8.2), with yield and ultimate strength values from the Nuclear Systems Materials Handbook (TID-26666). The above allowable pressures are conservative since the thickness of the cylindrical portion of the containment has increased subsequent to the calculations due to other design considerations.

Question CS220.35

a) Code Case N-284 (1980) should be referenced and applied as applicable.

b) The abscissa on Figures 3.8A-1, 3.8A-4, and 3.8A-6 should be labeled R/t and not R/1.

c) Both quadratic and linear interaction curves are used. Most authors recommend using linear interaction curves. Are the nonlinear interaction curves conservative?

d) The R/t range for the containment shell is in a borderline region where either elastic or plastic buckling could occur. For fabricated shells of this type, imperfections can greatly influence the elastic buckling loads. Also, plastic buckling can be influenced by large residual stresses that can be present. For these reasons the factors of safety given in Table 3.8A-2 seem to be low. Please justify these factors.

e) How will buckling be evaluated for dynamic loads?

Response:

a) See answer to Question 220.25. Code Case N-284 addresses containment shell buckling. This non-mandatory Code Case is a recent document which was not in existence at the time when the Containment Vessel design was initiated. The Project recognized the need for specific buckling criteria and an appropriate criterion was developed which is described in PSAR Appendix 3.8-A. While the quantitative effect on the design of using Code Case N-284 has not been determined, the intent of the buckling criteria used by the Project is similar to the Code Case, in that the criteria address buckling modes, provide capacity reduction factors and factors of safety, and similar interaction equations for buckling.

b) The typographical errors on Figures 3.8A-1, 3.8A-4 and 3.8A-6 have been corrected.

c) The selection of quadratic or linear interaction curves is dictated by loading combination. For example, interaction curves for loading combinations involving torsion are quadratic because modes shapes are dissimilar. Quadratic interaction curves are used in Code Case N-284. In the design buckling criteria, for each loading combination the shape of the nonlinear interaction curve is identical to that given in a widely

used shell design document, "Structural Analysis of Shells" by E. J. Baker, L. Kovalesky and F. L. Rish, McGraw-Hill, 1972. Therefore, it is concluded that each interaction curve is conservative.

d) The factors of safety given in Table 3.8A-2 are based on the requirements that the effects of initial imperfections and plasticity are adequately considered in the calculation of critical loads. These effects are accounted for by the coefficients C, K, and H in Figures 3.8A-1 to 3.8A-10. The effects of residual stresses on predicted critical loads are negligible for ring-stiffened cylinders subjected to axial compression (the most severe loading condition).

e) Buckling is evaluated for dynamic loads by comparing the peak dynamic stresses with the static critical (allowable) stresses.

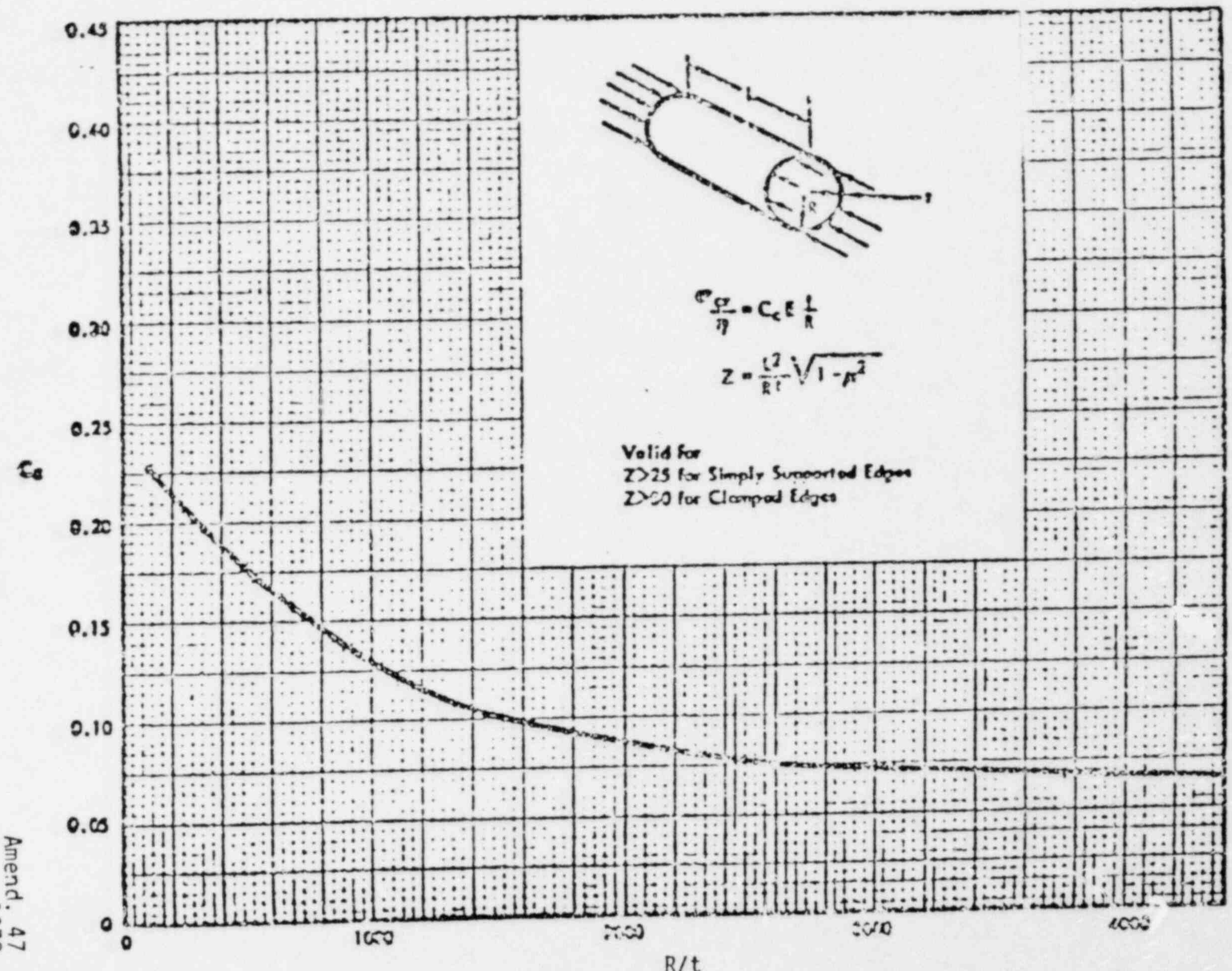

Figure 3.8A-1 Buckling-Stress Coefficient, C_C, For Unstiffened Unpressurized Circular Cylinders Subjected to Axial Compression

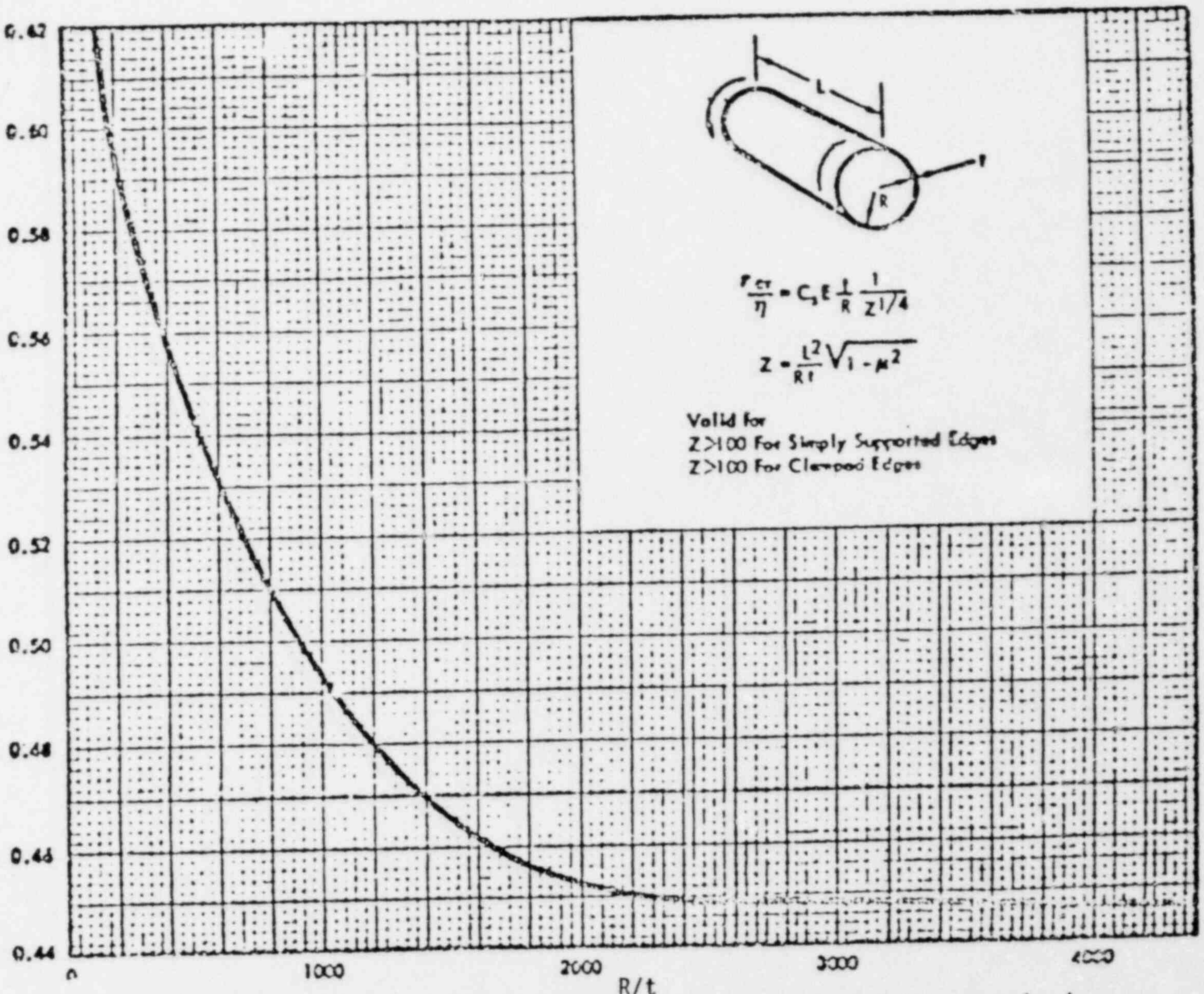

Figure 3.8A-4 Buckling-Stress Coefficients, C_B, For Unstiffened Unpressurized Circular Cylinders Subjected to Torsion

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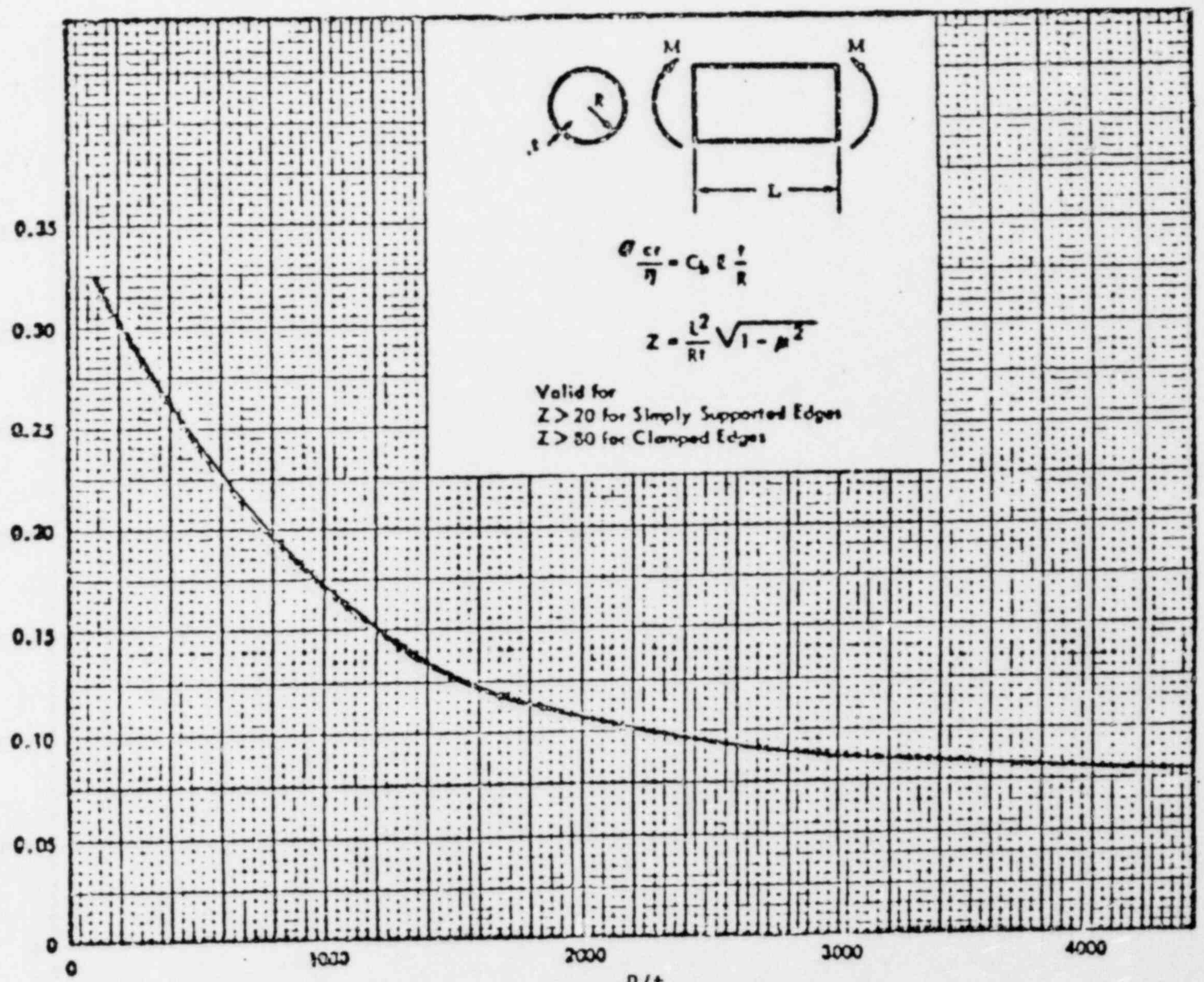

Figure 3.8A-6 Buckling-Stress Coefficient, C_6, For Unstiffened Unpressurized Circular Cylinders Subjected to Bending

Question CS250.1

In Section 3.2.2, "Safety Classifications," of the PSAR, the applicant stated compliance with the quality requirements of paragraph 50.55a, "Codes and Standards," of 10 CFR Part 50, except items (g), "Inservice Inspection Requirements," (i), "Fracture Toughness Requirements," and (j), power reactors for which a hearing notice on the application for a construction permit was published on or before December 31, 1970.

We concur with the applicant that the CRBRP is not subject to the conditions of paragraph 50.55a (g), (i), and (j) of 20 CFR Part 50. The conditions of paragraph 50.55a are applicable to boiling or pressurized water-cooled plants. However, pursuant to paragraph 50.55, "Conditions of construction Permits, "we require conformance to the Inservice Inspection and Fracture Toughness requirements of the CRBR Principal Design Criteria stated in the PSAR. Deviation from the CRBR Design Criteria should be identified and justified.

The Inservice Inspection and Fracture Toughness surveillance requirements to be specified for the CRBRP must provide an acceptable level of quality and safety in design and construction under operating, maintenance, testing and postulated accident conditions.

Response

I. Inservice Inspection - The Inservice Inspection plan for the Clinch River Breeder Reactor Plant is described in Appendix G of the PSAR. Any exceptions to the applicable Design Criteria are identified and justified in Appendix G. Also, see the response to Question CS250.2.

II. Material Surveillance

1. Reactor Coolant Boundary: CRBRP Criterion 30

   Provisions for material surveillance for the reactor coolant boundary are incorporated in the design of the reactor vessel internals and are documented as follows:

   a) The response to General Design Criterion 30 states that appropriate surveillance samples will be placed inside the reactor vessel thus providing means for monitoring and evaluating potential material degradations.

   b) A CRBRP Material Surveillance Program for the Reactor Coolant Boundary has been drafted, and provisions of this program have been incorporated in applicable specifications and other design documents. This program report is being finalized and a summary will be included in the FSAR.

2. Intermediate Coolant Boundary and Steam Generator System: CRBRP Criterion 33

The Intermediate Coolant Boundary and Steam Generator System surveillance program is being developed. A summary of this program will be included in the FSAR.

Question CS250.3

Identify the components and supports in the reactor coolant system and connecting systems (including the steam generator) which have been constructed, stating the purchase date and the Code, Standards, and criteria to which they were fabricated. Describe the procedures used for their storage. Indicate the difference in the purchase requirements and the Codes, Standards and criteria in effect at the present time. The use of the components should be justified on the basis that they will provide an equivalent degree of system integrity and safety as if fabricated to the requirements of the current Codes, Standards, and criteria.

Response

The following components have been procured and fabricated:

Component	Contract Date
Reactor Vessel	4/18/75
Closure Head	11/14/75
Guard Vessels	2/26/76
Core Support Structure	3/08/76
Intermediate Heat Exchangers	4/23/75
Primary Check Valves	11/75
Primary Control Rod Drive Mechanisms	5/30/75

Listing of the applicable Code editions, Code Cases, and RDT Standards are found in the PSAR as follows:

Components	Location
Reactor Vessel, Closure Head, and Guard Vessels	Table 5.2-1
Core Support Structure	Section 4.2.2.3
Primary CRDM	Section 4.2.3.1.5
PHTS Components	Section 5.3.1.2
IHTS Components	Section 5.4.1.1 and 5.4.1.2

The procedures used for storage of the identified components and supports implement the intent and guidance of ANSI N45.2.2 as endorsed by Regulatory Guide 1.38.

Each item important to the safe and reliable operation of the nuclear power plant is assigned by the designer, a classification level as defined in ANSI N45.2.2. The Project has established and maintains storage facilities which provide the levels of storage as defined in ANSI N45.2.2.

The Designer establishes packaging, handling, and storage requirements to be satisfied for each component which are based on the importance or complexity of the component, the classification of an available storage facility, and the component characterisitcs which must be protected or preserved (i.e., exterior surface, interior surface, contamination, temperature variations, moisture, etc.). The packaging, handling, and storage requirements include:

- o Receipt inspection criteria,
- o Unpacking and re-packing criteria,
- o Storage maintenance (purge, dessicant, rotation, lubrication, etc.) criteria, and
- o Storage maintenance verification criteria.

The procedures control both housekeeping of storage facilities per ANSI N45.2.2 and component maintenance per the packaging, handling, and storage requirements from the designer.

The current applicability of Codes and Standards used in early procurements of CRBRP components was addressed orally as part of the May 6-7, 1982, presentation to the NRC on High Temperature Design. The documentation requested will be provided in Reference QCS210.1-1.

Question CS410.2 (9.1.4)

It is our position that overhead cranes whose failure could damage spent fuel or essential equipment be designed such that in the event of the SSE they can retain control of and hold their load. The bridge and trolley should be designed to remain in place on their respective runways with their wheels prevented from leaving the tracks during the SSE. The bridge should remain on the runway with brakes applied, and the trolley should remain on the crane girders with brakes applied. The crane should be designed and constructed in accordance with Regulatory Position 2 of Regulatory Guide 1.29, "Seismic Design Classification." The maximum critical load plus operational and seismically induced pendulum and swinging load effects on the crane should be considered in the design of the trolley, and they should be added to the trolley weight for the design of the bridge.

For the polar crane, cask handling crane, and other cranes whose failure could damage spent fuel or essential equipment, demonstrate that you meet this position.

Response

The following responses apply to the RCB Polar Crane, and the RSB Bridge Crane (cask handling crane) and the SGB Gantry Crane:

(1) For critical loads, overhead cranes are designed such that the SSE will not result in dropping or losing control of the heaviest critical load. The cranes are designed as Seismic Category I, single failure proof, redundant cranes to meet NUREG-0554.

(2) All cranes are provided with seismic restraints such that the bridge and trolley will remain in place on their respective runways during the SSE. The bridge will remain on the runway with brakes applied and the trolley will remain on the crane girders with brakes applied.

(3) The subject cranes are designed in accordance with Regulatory Position 2 of Regulatory Guide 1.29, "Seismic Design Classification."

(4) Operationally and seismically induced pendulum (i.e. swinging) load effects have been incorporated in the crane designs by adding these loads to the maximum critical load and dead load.

Question CS410.3

Provide the results of an analysis which demonstrates that spent fuel and essential equipment will not be damaged by a heavy load drop due to a handling system malfunction. Include consideration of cask handling crane failure resulting in a cask drop, polar crane failure resulting in dropping the heaviest load handled by this crane over essential equipment including the reactor vessel, and other pertinent potential handling system malfunctions. The analysis should satisfy the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," Section 5 and Appendix A with due consideration for the differences between CRBR and LWR design.

Response

The CRBRP approach to control of handling heavy loads is consistent with the guidelines of NUREG-0612. The approach takes advantage of a fundamental difference between CRBRP and LWR designs, i.e., reactor refueling and spent fuel storage fuel handling are done "through-the-head" rather than "open-head" or "open-pool."

Selected heavy loads use the single-failure-proof cranes in the RSB and RCB. In addition to the use of single-failure-proof cranes and associated lifting devices with the following features or procedures are used.

- o Identification of safe load paths,
- o Administrative controls over handling of heavy loads over or near the reactor, Ex-Vessel Storage tank of Fuel Handling Cell,
- o Polar crane interlocks to prevent handling of heavy loads and polar crane operations over or near the reactor during reactor operation, and
- o Design and analysis of the reactor and fuel storage facilities for the impact of the heaviest anticipated load carried by the single-failure-proof main hooks of the RSB and RCB cranes (at the maximum lowering speed). This is over and above the requirements imposed by NUREG-0612.

Other heavy loads use the auxiliary non-single-failure proof cranes and associated lifting devices. For the EVST, FHC and Reactor, the effects of a load drop of the maximum weight from the highest administratively allowed working height are analyzed. Due to a polar crane interlock, and identification of safe load paths, the drop from the auxiliary hook is analyzed only for reactor shutdown conditions. In these three locations, the postulated load drops do not affect criticality, damage the integrity of the liquid coolant boundary, or affect shutdown equipment.

The results from impact load analyses for the reactor head and fuel handling facilities are summarized below.

IMPACT LOAD ANALYSIS SUMMARY

Location	Pertinent PSAR Sections	Impact Loading	Consequences
Ex-Vessel Storage Tank	9.1.2.1.1	Load lowered by a single failure proof crane	No damage
		Load drop from non-single failure proof crane	Stresses less than ASME Code allowables for no gas leakage
Fuel Handling Cell	9.1.4.10.1	Load lowered by a single failure proof crane	No damage
		Load drop from non-single failure proof crane	Damage factor of 4 for reinforced concrete. No structural failure. Radiological consequences of possible leakage of FHC atmosphere enveloped by reactor cover gas leak (see PSAR 15.5.2.4).
Reactor	15.5.2.5.1	Load lowered by a single failure proof crane	Possible damage to head mounted components. Radiological consequences of cover gas release are enveloped by a discussion in PSAR Section 15.5.2.4 due to nature of crushed CRDMs. This event is not as severe as the reactor cover gas release event because the seal failure would result in a more gradual leakage of cover gas.

The procedures and identification of non-critical loads to be lifted over the reactor closure head during reactor shutdown using the non-single failure-proof-crane are currently under development.

The Spent Fuel Shipping Cask (SFSC) is required to have single-failure-proof lifting devices and can only be handled by the single failure proof Reactor Service Building (RSB) overhead crane main hook. In addition, a safe load path has been identified for the SFSC that does not pass over or near the EVST, FHC, or safety-related equipment.

The radiological impact of the drop of the SFSC from the single-failure-proof crane onto the SFSC handling shaft is provided in Section 15.7.3.2.

The effect of heavy loads dropped on safe shutdown equipment in the RCB has been analyzed. Any heavy load dropped from the auxiliary hook can only affect one channel of the shutdown system. Since the Reactor Shutdown System is designed to IEEE 379, (Application of Single Failure Criterion to Nuclear Protection Systems), as described in PSAR Section 7.2.2, failure of that one channel, by damage to sensors, cabling, or transmitters, will not preclude operation of sufficient equipment to achieve a safe shutdown.

Question CS410.19 (9.7.3)

The normal and emergency chilled water systems provide cooling for plant HVAC systems. HVAC units serving areas containing sodium or NaK are provided with drains to carry away chilled water leakage to prevent moisture carry-over in the HVAC ducting. Leak detectors are provided in the drains to detect chilled water system coil failure. Activation of the detector results in automatic closure of the chilled water coil isolation valves. Justify the use of non-safety related normal chilled water system piping and valves in HVAC units serving areas containing sodium and NaK.

Response

With the exception of the SGB loop cells, the HVAC units provided with normal chilled water and serving areas containing sodium and NaK are located outside the sodium and NaK cells. These cells do not require safety-related cooling. Accordingly, their associated HVAC units are classified as non-safety related. For the SGB loop cells. Three barriers between the sodium and water are provided as follows:

a) Chilled water piping walls

b) HVAC equipment walls which serve as spray shields

c) Sodium piping walls

The safety classification of these barriers is currently being evaluated to determine if they provide adequate protection against a sodium/water reaction. The results of this evaluation will be provided in a future ammendment.

<u>Question 421.01</u>

During meetings with the applicant and Westinghouse, several discussions have been held concerning the fact that the primary and secondary shutdown systems do not each, individually, comply with Section 4.7.3 of IEEE-279 on Control and Protection System Interaction. The applicant should document for inclusion in the PSAR the justification of the adequacy of the proposed design for complying with Section 4.7.3 of IEEE-279. This justification should include a discussion of the system adequacy with respect to control and protection system interaction during periodic testing of a protection system channel or when a protection system channel is out of service for maintenance. If the justification includes the use of a median selector for control signals, plans to periodically test the median selector during plant operation should also be discussed.

<u>Answer</u>

The two PPS reactor shutdown systems, considered together, always meet Section 4.7.3 of IEEE-279.

The provision of two diverse and independent shutdown systems provides the plant with an unusually high degree of protection against common mode failure incidents.

Under normal operating conditions, and in the great majority of abnormal conditions, each system separately fully meets the requirements of IEEE-279.

However, in a limited number of situations, it is possible that during testing or maintenance of a channel which supplies signals to both protection and control, a single sensor failure could initiate plant control actions requiring protective action and simultaneously prevent proper protective channel response. Applying the criteria of Section 4.7.3 of IEEE 279, which requires the assumption of a second random failure for these channels, results in the assumption that the protective function in one of the systems would be disabled. In these situations, the redundant functions in the alternate system always provide protection as required by Section 4.7.3 of IEEE-279.

The likelihood of such potentially disabling situations is limited by the use in the control system of a median select arrangement which precludes response to abnormally high or low signals. However, when one protection channel is placed in the trip condition, a simultaneous sensor failure in a low direction could potentially result in disabling the protection system at the same time that a faulted control system calls for a power increase.

With regard to validation of the median selector, most failures of the median selector are self-annunciating (i.e., abnormal plant response is detected by the operator or subsystems within both Shutdown Systems leading to a plant trip). Other median selector failures (eg., output not following the true median input) occurring together with a sensor failure do not cause a plant response. In any case, median selector failures do not invalidate the performance of the related protection system. Nevertheless, functional testing of the median selector circuits is performed annually as part of the scheduled maintenance.

In the majority of situations, further built-in control features (such as built-in overall temperature loops or rod movement blocks related to alternate flux measurements) will prevent any such overpower conditions from developing.

Where this is not the case, as previously noted, protection is provided by the second independent protection system.

Section 4.7 of IEEE 279 deals with control/protection system interaction. There is no control/protection system interaction for the SCRS.

Question CS421.5

Section 7.6 of the PSAR states that the Radiation Monitoring System contains safety related components which are discussed in Chapter 11. However, Chapter 11 does not discuss these safety related components. Correct the PSAR to identify the safety related compoents of the Radiation Monitoring System.

Response

A PSAR amendment will be submitted by July 1982 which will identify safety related components of the Radiation Monitoring System. Safety-related process and effluent monitors will be described in Section 11.4.2.2.8 and identified in Table 11.4-1. High range containment area monitors which are safety related will be identified in Section 12.1.4.1.

Question 421.08

Document the design provisions for conducting response time tests of BOP and NSSS protection systems in accordance with R.G. 1.118. Identify safety-related systems that do not have provisions for response time testing. Discuss the techniques to be used to periodically measure safety-related sensor time responses.

Response:

The CRBRP Protection systems are being designed to comply with R.G. 1.118. The NSSS protective systems (RSS, CIS and SGAHRS Initiation) use an overlap testing technique to verify that system response times are within acceptable limits. The following features are provided for response time testing:

A. For the neutron detectors, detector-cable capacitance checks are made with the detectors and cables in the as-installed configuration to identify increases in capacitance which could affect time response of the channel.

B. For the reactor vessel level instruments, the sensors are checked in the as-installed configuration by inserting a step increase in the primary coil excitation current at a steady state sodium temperature and level and monitoring the increase in secondary coil output voltage due to the increase in magnetic field strength.

C. For channel response time testing, not including sensors, a test signal source is connected to the channel which can simulate the sensor input over the entire range. Measurements of channel input changes and channel output changes are recorded. These tests include istrument channel equipment and the comparator trip outputs to the logic trains. Where protective functions use two or more variables, channel time response is determined for each input to the function one variable at a time. The remaining variables are adjusted to a conservative value within the normal operating range.

D. The Primary RSS and Steam Generator Auxiliary Heat Removal System (SGAHRS) initiating logic time response is checked during functional testing of the logic. Pulses are inserted into the redundant instrument channel comparators to simulate the eight possible combinations of trip and reset. For test inputs with two or more redundant channels tripped, the logic output is checked for a trip condition. This trip output must be detected within the propagation delay limits of the logic train or it is flagged by the tester as a failure.

E. The Primary RSS scram breaker time response is checked by inserting trip signals to the logic train and monitoring the time to current interruption on the output side of the breakers. Each breaker is tested separately to assure compliance with response time requirements.

F. The Secondary RSS logic is tested by inserting a trip input to the comparator and monitoring the time required until the current to the solenoid scram valve in interrupted.

G. Time response testing of the Secondary RSS scram solenoid valves is included with testing of the Secondary rods.

H. The CIS logic and breakers are tested in a similar fashion to D and E.

I. Response time measurement of pressure or differential pressure is performed by applying a pulse with a specific ramp rate and comparing the output of a tested transducer with the output of a fast acting reference transducer. The output of both transducers is recorded on a strip chart for subsequent computation of the response time.

J. Response time measurement of thermocouples or RTDs is facilitated with the use of a loop current step response analyzer. The loop current step response analyzer sends a pulse of current to the thermocouple or RTD. The analyzer receives time varying voltage data from the temperature sensing device, digitizes and stores test data. The analyzer then computes the sensor time constant and displays this on a panel meter.

K. A sweep generator shall be provided for measuring pump speed signal conditioning time response. The sweep generator shall have an auxiliary voltage output that is proportional to frequency output. The sweep generator shall supply an input to the pump speed signal conditioning modules and the proportional voltage output will be supplied to an oscilloscope. The output of the pump speed signal conditioning modules will also be supplied to the oscillograph.

L. Response time measurement of the sodim flow signal conditioning modules shall be accomplished utilizing a millivolt generator that provides a ramp output. The methodology used is analagous to the methodology described in the previous paragraph.

M. A sensor response time monitor will be used to determine if time response degradation has occurred in PPS channels. The device samples the normal fluctuations around the average valve of the sensor output and determines the time response characteristics of the sensor based on the number of times the output signal crosses its average valve in a fixed time interval. This device may be used on RTDs, thermocouples, pressure sensors, level sensors, and flow sensors as a method of determining time response degradation of PPS channels. Use of this methodology is fast and efficient but cannot be used for initial time response measurements. Only time response degradation can be measured.

N. A test and calibration signal source is permanently installed on each PPS Shutdown Panel. The signal source is used to inject a step test signal for comparater response time measurement. The step test signal is inserted into the channel to cause a PPS comparator to trip The output of the comparator is compared to the output of the test signal and time response characteristics are calculated.

O. SGB flooding is detected with temperatures and humidity sensors. When flooding is detected the feedwater isolation valves shut to prevent damage to SGAHRS components. A method of verifying humidity sensor time response characteristics is to be determined.

P. Radiation monitors are provided with a built-in check source. The check source is normally shielded from the detector, and the shield is solenoid operated. Response time can be measured by electrically actuating the solenoid operated shield, and obsreving the monitor output, taking due account for solenoid response time.

Q. Gas detectors can be tested similarly to pressure instruments by injecting known composition of gas through the test valves and observing the instrument output.

R. Flow sensors whose operation is based on the hot wire technique are not tested in situ. They would be removed from the pipe or duct and placed in a test fixture in which flow can be suddenly terminated, and the flow sensor output observed.

S. Sound powered phone jacks, test cabling, and test points are incorporated into the design to facilitate the testing methods described above. Most of the testing hardware is portable but some permanently mounted equipment is provided to minimize the effort required to measure the time response of the PPS comparaters.

QUESTION CS 421.09

Identify where instrument sensors or transmitters supplying information to more than one protection channel, to both a protection channel and control channel, or to more than one control channel, are located in a common instrument line or connected to a common instrument tap. The intent of this item is to ve fy that a single failure in a cmmon instrument line or tap (such as break or blockage) cannot defeat required protection system redundancy.

RESPONSE

Instrumentation sensors or transmitters located in instrument lines or connected to instrument taps do not supply more than one protection channel or control channel. Therefore, the required protection action will not be defeated by a blockage or breakage of an instrument line or instrument tap.

However, there are instrument sensors and transmitters located in instrument lines or connected to instrument taps which provide signals to both a protection channel and a control channel. In all cases, both the protection and control function has a three channel input redundancy. These redundant channels use separate instrument lines and taps. For example, the superheater steam flow Venturi provides three separate taps located 120^{o} apart for the redundant sensors. Thus, a single failure resulting from a blockage or a breakage in an instrument line or tap will not defeat the required protective action.

Response to NRC Question 421.09

In general the instrumentation shall meet the criteria of IEEE-279-1971 and RDT Standard C16-1T-1969, also the single failure criteria of IEEE 379-1972 and Regulatory Guide 1.53.

A. Pertaining to the Plant Protection System (PPS) channels, each reactor shutdown system (RSS) shall use three redundant channels of instrumentation to measure plant parameters.

B. Accident monitoring equipment redundance is per WARD-D-0307.

C. For Class 1E equipment other than A&B, redundancy will be accomplished by one of the following criteria.

   1. Redundant process loops.

   2. Redundant instrument channels.

The redundancy requirements of the instrumentation applicable to this question are satisfied as follows:

- Three redundant channels for each loop are provided for the following PPS channels:

   1. Feedwater flow
   2. Feedwater temperature
   3. Superheater steam flow
   4. Superheater steam temperature,
   5. Superheater steam pressure
   6. Reactor products vent flow (east evaporator, west evaporator, superheater, IHTS Na expansion tank.

   Two auxiliary feedwater flowmeter transducers are provided for each AFW loop to insure isolation capability with loss of one 1-E power division (each transducer pair is powered from different 1-E power sources).

Question CS421.14

Discuss the CRBR design pertaining to bypassed and inoperable status indication. As a minimum, provide information to describe:

1. Means to be used for compliance with the recommendations of R.G. 1.47.

2. The design philosophy to be used in the selection of equipment/systems to be monitored.

3. How the design of the bypass and inoperable status indication systems will comply with positions B1 through B6 of ICSB Branch Technical Position No. 21.

The design philosophy should describe as a minimum the criteria to be employed in the display of inter-relationships and dependencies on equipment/systems and should insure that bypassing or deliberately induced inoperability of any auxiliary or support system will automatically indicate all safety systems affected.

Response

The following responds to Items 1, 2 and 3:

1. A discussion regarding compliance with R.G. 1.47 is provided in new PSAR section 7.5.12.

2. The safety functons and systems to be monitored by Inoperable Status Monitoring System (ISMS) are based upon Engineered Safety Features of Chapter 6 of the PSAR and are shown in the Table 7.5-4 of PSAR Section 7.5. There are 2 active safety systems which are not included in ISMS: the Reactor Shutdown System, which provides separate indication, and the Containment Isolation System, for which no bypasses or deliberately induced inoperable states have been identified. (Reactor Shutdown System bypasses are discussed in Section 7.2.1.1 of the PSAR.) The design of ISMS will employ the following steps to ensure that the system inter-relationships and dependencies on auxiliary systems are properly identified.

   - For each system and subsystem in the attached table, the states of the components which lead to system inoperability will be identified.

   - The Maintenance Outline Procedures and Operating Outline Procedures will be reviewed to identify any maintenance testing, or surveillance activities which would cause any active component to be bypassed or inoperable.

   - The results of the first two items will be combined to define the components to be monitored and to develop the logic for identifying inoperable systems per Regulatory Guide 1.47.

- o In addition, the auxiliary and support systems required for operation of all active components in the safety systems will be identified and the above 3 steps repeated for these identified auxiliary and support systems. Inoperability of these auxiliary and support systems would be indicated by auxiliary system inoperability indications and an indication of the inoperability of the affected components.

3. The design of the CRBRP bypass and inoperable status indication systems is intended to comply with positions B1 through B6 of ICSB Branch Technical Position No. 21 as follows:

BTP1. ["The bypass indicators should be arranged to enable the operator to determine the status of each safety system and determine whether continued reactor operation is permissable."]

Bypass indication for safety systems is to be combined on a single ISMS indicator panel with separate indications for each of the following subsystems: (Ref. PSAR Sec. 7.5.12)

- o Decay Heat Removal System
- o Fuel Storage Heat Removal System
- o Control Room Habitability
- o Annulus Filtration
- o Reactor Service Building (RSB) Filtration

These dedicated indicators are activated whenever a system is determined bypassed or inoperative.

In addition, the ISMS is supported by Plant Annunciator System (PAS) and the Plant Data Handling and Display System (PDH&DS), and changes in the safety system status are transmitted to the PAS for audible and visual annunciation to the operator. PDH&DS cathode ray tube (CRT) displays may be used to provide the operator information about safety systems.

BTP2. ["When a protective function of a shared system can be bypassed, indication of that bypass condition should be provided in the control room of each affected unit."]

CRBRP shares no safety system with other units.

BTP3. ["Means by which the operator can cancel erroneous bypass indications, if provided, should be justified by demonstrating that the postulated cases of erroneous indications cannot be eliminated by another practical design."]

Activation of bypass indication is provided by a computer program which is not accessible to the plant operator. Cancellation of bypass indication is normally only possible through removal of the condition which caused the bypass indication (e.g., reclosing of a critical valve or breaker). If the condition is erroneous, the cause of the error (e.g., a short-circuited wire) must be determined and corrected in order to cancel the bypass indication.

BTP4. ["Unless the indication system is designed in conformance with criteria established for safety systems, it should not be used to perform functions that are essential to safety. Administrative procedures should not require immediate operator action based solely on the bypass indications."]

The CRBRP bypass and inoperable status indication system is not used to perform functions essential to safety. CRBRP operating procedures will not require immediate action in response to bypass indications.

BTP5. ["The indication system should be designed and installed in a manner which precludes the possibility of adverse effects on plant safety systems. Failure or bypass of a protective function should not be a credible consequence of failures occurring in the indication equipment, and the bypass indication should not reduce the required independence between redundant safety systems."]

The ISMS equipment shall be isolated from the associated safety related equipment so as to preclude any abnormal or normal action of the ISMS from preventing the performance of a safety function. It is intended that all electrical input connections to ISMS from safety related equipment are electrically isolated at the safety related equipment.

BTP6. ["The indication system should include a capability of assuring its operable status during normal plant operation to the extent that the indicating and annunciating function can be verified."]

The ISMS will provide a means of verifying that the indicator lights are functional.

7.5.12 Inoperable Status Monitoring System

7.5.12.1 Design Description

The Inoperable Status Monitoring System (ISMS) provides for an automatic indication at the system level of the bypassed or deliberately induced inoperability of selected safety systems. The indication of inoperative and bypassed status will be provided in conformance with Regulatory Guide 1.47, "Bypassed and Inoperative Status Indication for Nuclear Power Plant Safety Systems". The Reactor Shutdown System (RSS) includes indication of bypassed status and is not part of ISMS. See Section 7.2 for the RSS details.

Specific capabilities of the ISMS include:

- Monitoring of safety system variables and alerting the unit operator visually and audibly of a bypassed or deliberately induced inoperable safety system.
- A manual capability to activate the safety system indicators.

No direct safety or protection function is performed by the ISMS. The list of status information and monitoring provided by ISMS is provided in Table 7.5-4. The data acquisition and processing components of the ISMS are located in the control room area behind the Main Control Panel (MCP). The unit operator interface controls and indicators are located so they can be seen from the MCP. The ISMS data acquisition system obtains the sensor status information processes the sensor data, and provides the status indication to the unit operator. The processed data is transmitted to the Plant Data Handling & Display System (PDH&DS) for presentation via cathode ray tube (CRT) displays in support of ISMS.

7.5.12.2 Design Analysis

The ISMS is completely designed to conform to the requirements of Regulatory Guide 1.47, "Bypassed and Indication for Nuclear Power Plant Safety Systems". The ISMS in combination with the bypass status indication portions of the Reactor Shutdown System provides the complete system level coverage of safety system indication and manual activation of indicators required by Regulatory Guide 1.47.

The safety systems provide the isolation devices for associated safety-related equipment so as to preclude any action of the ISMS from preventing the performance of a safety function.

Table 7.5-4

Safety Functions and Primary Systems Monitored by ISMS

SAFETY FUNCTION MONITORED	SYSTEMS/SUBSYSTEMS MONITORED	
Decay Heat Removal	SGAHRS	PACC
		Aux FW/Vent
		Steam Generator
		Heat Transport
	DHRS	PHTS Heat Transport
		Direct Heat Removal System
Fuel Storage Heat Removal	EVST Cooling	Forced Circulation
		Natural Circulation
Control Room Habitability	Control Room Filtration H&V	
Annulus Filtration	RCB/Annulus H&V	
RSB Filtration	RSB H&V	

Question 421.17

The information supplied for remote shutdown (PSAR Section 7.4.3) from outside the control room is insufficient. Therefore, provide further discussion to describe the capability of achieving hot or cold shutdown from outside the control room. As a minimum, provide the following information:

a) A table listing the controls and display instrumentation required for hot and cold shutdown from outside the control room. Identify the train assignments for the safety related equipment.

b) Design basis for selection of instrumentation and control equipment on the hot shutdown panel.

c) Location of transfer switches and the remote control station.

d) Description of transfer switches and the remote control station.

e) Description of isolation, separation and transfer/override provisions. This should include the design basis for preventing electrical interaction between the control room and remote shutdown equipment.

f) Description of control room annunciation of remote control or overridden status of devices under local control.

g) Description of compliance with the staff's Remote Shutdown Panel position.

Response

The response to this question is provided in the amended text for Section 7.4.3.

7.4.3 Remote Shutdown System

7.4.3.1 Design Description

7.4.3.1.1 Function

The Remote Shutdown System provides the means by which (1) safe shutdown conditions of the reactor plant can be established and maintained from locations outside of the Control Room in the event that the Control Room must be vacated; (2) hot shutdown conditions can be achieved and maintained; and, (3) if desired, the plant can be cooled to and maintained at the refueling temperature.

7.4.3.1.2 Design Basis

The Remote Shutdown system is designed to use equipment located outside of the Control Room to place the reactor and plant into a safe shutdown condition under the following conditions:

(a) The evacuation of the Control Room is not coincident with any other abnormal plant condition with the one exception that loss of offsite power may occur.

(b) No severe natural phenomena such as earthquake, tornadoes, hurricanes, floods, tsunami and seiches (from 10CFR50, Appendix A, Criterion 2) occur coincidently with the excavation of the Control Room.

(c) The plant remains in an orderly shutdown status from the initiation of the evacuation of the Control Room to the time that command of the shutdown is re-established outside of the Control Room.

(d) The remote shutdown operations will be commanded from one location and will use plant systems operated in their local mode to effect the shutdown and decay heat removal.

(e) Plant instrumentation and control systems required for remote shutdown operations will have transfer switches located at the local panels to permit the plant operating personnel to select to operate from the local panels while isolating the remote controls or, conversely, to operate from the control room while isolating the local controls. The transfer of control of a plant system from the remote to the local mode is annunciated in the control room.

(f) Communications between the Remote Shutdown Monitoring Panel (RSMP), the command location for remote shutdown operations, and the SGAHRS panels and other local panels during remote shutdown operations will be by the Maintenance Communication Jacking (MCJ) system utilizing a sound-powered telephone.

7.4.3.1.3 Remote Shutdown Operations

The RSMP will be located in Cell 271 of the 836'-0" level of the SGB. The RSMP will have indications (see 7.4.3.1.4) from which an operator can assess the progress of the shutdown, and it will be the location from which that operator will command the operation of the plant systems being operated in their local mode to effect shutdown.

The Division 1, 11 and 111 SGAHRS (Section 7.4.1) local panels will be located in Cells 272A, B and C respectively, in close proximity to the RSMP, on the 836'-0" level of the SGB-1B. The SGAHRS, operated in its local mode, will be used to control the removal of heat from the reactor plant to achieve and stabilize the plant at the desired plant temperature (hot shutdown or refueling temperature). The local SGAHRS panels will have all of the controls and indications necessary to completely control the system. All signals from the Control Room to the SGAHRS panels are buffered to prevent faults occuring in the Control Room from propogating back to the SGAHRS panels. All SGAHRS component controls can be transferred to local at the local SGAHRS panels. Placing the transfer switches in "local" overrides all control functions in the Control Room.

The Division 1, 11 and 111 OSIS local panels are located in SGB Cells 272A, B and C with the SGAHRS panels, and will be operated in the local mode when required to control heat removal from the plant in conjunction with the operation of SGAHRS. Isolation of OSIS panel controls from the Control Room is incorporated in the design. Steam drum drain and superheater outlet isolation valve controls can be transferred to local at the local OSIS panels.

Whenever any SGAHRS component control transfer switch is placed in the "local" position an alarm is initiated in the Control Room to alert the Control Room operator. The same statement is true for the steam drum drain controls and superheat outlet isolation valve controls on the OSIS panels.

If offsite power is lost coincident with having to achieve a safe shutdown condition in the reactor plant from outside of the Control Room, the diesel generators will start and function in accordance with the design provided by the Building Electrical Power System. Any operator actions required in conjunction with operating and loading the diesel generators will be done in the local operating mode at the DG local panels.

In the event that the Control Room must be vacated, reactor scram and SGAHRS operation will be initiated manually. The operating personnel will move to the 836'-0" level of the SGB where the SGAHRS in the local mode will effect heat removal and stabilization of the plant temperatures. Operation of the SGAHRS in the local mode will effect heat removal and stabilization of the plant temperatures. The plant shutdown will be directed by the operator at the RSMP who will also assign operating personnel not continuously occupied in operating SGAHRS to oversee or operate other systems as required.

Movement of personnel within the plant and access to building cells and local panels will be controlled by the facilities and procedures of the Industrial Security System.

7.4.3.1.4 Equipment Design

The RSMP is the only piece of equipment provided by the Remote Shutdown System. It will be a vertical sided cabinet assembly containing meters and a phone jack panel. The meters will receive buffered signals from the initiating systems and, thus, do not require transfer switches to isolate them from the Control Room. The phone jack panel will permit the operator at the RSMP to communicate with the five NSSS or Nuclear Island buildings by means of any of the three MCJ circuits provided in each of the buildings. In addition, communications among the buildings can be established through the phone jack panel on the RSMP.

The indications provided on the RSMP are as follows:

o For each primary heat transport system loope,

1 - Pump outlet sodium temperature indication	(3 total)
1 - Reactor inlet sodium temperature indication	(3 total)
1 - Sodium pump shaft speed indication	(3 total)

o For each intermediate heat transport system loop,

1 - IHX outlet sodium temperature indication	(3 total)
1 - IHX inlet sodium temperature indication	(3 total)
1 - Sodium pump shaft speed indication	(3 total)

o For each superheated steam loop,

1 - Temperature indication	(3 total)
1 - Steam flow indication	(3 total)

o One reactor vessel sodium level meter (long probe)

o For each Diesel Generator (3 total)

1 - Wattmeter
1 - Frequency meter
1 - Varmeter
1 - Voltmeter with phase selector switch
1 - Ammeter with phase selector switch

In addition to the foregoing indications, other indications used during remote shutdown operations that are not on the RSMP will be available as follows:

o <u>SGAHRS</u>

Controls and indicators used for the operation of each SGAHRS division are located on the three separate SGAHRS panels in cells 272A, B, and C. Each SGAHRS division is separate and redundant from the other divisions. See the response to question 421.04 for additional information about SGAHRS division assignments.

The following controls, indicators and alarms are on each SGAHRS panel.*

Controllers

Auxiliary Feedwater Flow
AFW Steam Turbine Steam Inlet Pressure
PACC Inlet Louver Position
PACC Fan Blade Position
Steam Drum Level
Steam Drum Vent
Superheater Vent

Analog Indicators

Protected Water Storage Tank Level
Protected Water Storage Tank Temperature
Auxiliary Feedwater Flow
Auxiliary Feedwater Pump Discharge Pressure
Steam Driven Turbine Steam Inlet Pressure
Steam Driven Turbine Speed
PACC Outlet Air Temperature
PACC Outlet Water Flow and Temperature
PACC Inlet Louver Position
PACC Fan Blade Pitch Position
Steam Drum Pressure and Water Level

Annunciators

Protected Water Storage Tank Level
PWST Temperature
AFW Supply Temperature
Steam Driven Turbine Speed
Driven Turbine Steam Inlet Pressure
Steam Driven Turbine Bearing and Lube Oil Temperature
High Motor Bearing Temperatures
SGAHRS Initiation

o Diesel speed and fuel oil indications will be available at the diesel generator local control panels in the Diesel Generator Building Cells 511 and 512.

*Each indicator, alarm and controller is repeated on each of the SGAHRS panels except for those associated with the AFW pumps. Panels A and B have the controls, alarms and indicators for motor driven AFW pumps A and B; Panel B has those associated with the steam driven AFW pump.

7.4.3.2 Design Analysis

The Remote Shutdown System provides the RSMP from which an operator can assess the progress of the plant shutdown and command the local operation of the plant systems (primarily SGAHRS) to effect the shutdown. It should be noted that the PACC subsystem of SGAHRS is automatically initiated by all reactor trips, and it remains in operation for the duration of the plant shutdown or as long as the reactor generates significant decay heat.

The Remote Shutdown System imposes no special requirements on the plant systems, but takes advantage of the following system design features:

o The ability to operate in both local and remote modes with isolation from and annunciation in the Control Room when operating in the local mode.

o The redundancy diversity, separation, isolation and reliability of the safety grade systems.

o The design and location of safety grade systems equipment that minimize the probability and effect of fires and explosions on the ability of the systems to perform their safety function.

o The redundant safety grade SGAHRS provides the capability to achieve and maintain hot shutdown and, if desired, to cool the plant to and maintain the plant at refueling conditions.

o When transferring SGAHRS to the local mode, the operator manually starts SGAHRS. Once started, SGAHRS automatically controls those parameters used to remove decay heat.

Question CS421.18

Provide documentation that verifies that control provided for safe shutdown from outside the control room will include the capability for reset of any engineered safety features equipment having a high likelihood of being automatically initiated during the normal transient occurring following a manual reactor trip. For example, the auxiliary feedwater system may be in this category.

Response

The Auxiliary Feedwater (AFW) and Protected Air Cooled Condenser (PACC) are subsystems of the Steam Generator Auxiliary Heat Removal System (SGAHRS). The AFW Subsystem is not initiated during the normal transient occurring following a manual reactor trip.

The PACC subsystem is automatically initiated by all reactor trips, and it remains in operation for the duration of the plant shutdown or as long as the reactor generates significant decay heat. The PACC has the capability of being reset at the local panels. Then, the operator can manually start and stop the PACC units. Once started the PACC units will automatically control steam drum pressure the same as in the main control room.

Question CS421.19

A number of concerns have been expressed regarding the adequacy of safety systems in mitigation of the kinds of control system failures that could actually occur at nuclear plants, as opposed to those analyzed in PSAR Chapter 15 safety analyses. Although the Chapter 15 analyses are based on conservative assumptions regarding failures of single control systems, systematic reviews have not been reported to demonstrate that multiple control system failures beyond the Chapter 15 analyses could not occur because of single events. Among the types of events that could initiate such multiple failures, the most significant are in our judgement those resulting from failure or malfunction of power supplies or sensors common to two or more control systems.

To provide assurance that the design basis event analyses adequately bound multiple control system failures you are requested to provide the following information:

1) Identify those control systems whose failure or malfunction could seriously impact plant safety.

2) Indicate which, if any, of the control systems identified in (1) receive power from common power sources. The power sources considered should include all power sources whose failure or malfunction could lead to failure or malfunction of more than one control system and should extend to the effects of cascading power losses due to the failure of higher level distribution panels and load centers.

3) Indicate which, if any, of the control systems identified in (1) receive input signals from common sensors, common hydraulic headers, or common impulse lines.

The PSAR should verify that the design criteria for the control systems will be such that simultaneous malfunctions of control systems which could result from failure of a power source, sensor, or sensor impulse line supplying power or signals to more than one control system will be bounded by the analysis of anticipated operational occurrences in Chapter 15 of the Final Safety Analysis Report.

Response

The design criteria for the Plant Protection System prohibits control system malfunctions from endangering plant safety. Therefore, there are no control system failures or malfunctions that seriously impact plant safety because of protection provided by the Plant Protection System (PPS). Failure in the following control systems could, however, cause a reactor scram to occur: Supervisory Control, Reactor Control, PHTS and IHTS Sodium Flow Control, PHTS and IHTS Pump Speed Control, Drum Level Control and Turbine Control. The Chapter 15 analysis envelopes the failure of multiple control systems due to loss of power since:

1) For loss of offsite power, the PPS trips the control rods upon loss of power to the sodium pumps. Action of the control system is irrelevant.

2) Primary rod control has redundant MG sets powered from non-UPS normal A and B sources. Loss of A or B does not affect rod motion. For loss of A and B, a PPS trip occurs due to steam/feedwater mismatch resulting from a turbine/generator trip.

3) Failure of electrical power (non-UPS normal A) to the Supervisory Control and Reactor Control Systems will not result in primary control rod withdrawal. The control rod rate circuit will produce a zero rod rate signal with zero power available. The worse that can happen on the loss of non-UPS normal electrical power is a reduction in coolant flow which is enveloped in the Chapter 15 analysis.

4) For Supervisory Control, Reactor Control, PHTS Sodium Flow control and IHTS Sodium Flow Control, the design provides for controllers in different cabinets each with redundant power supplies to eliminate power supply failures affecting several controllers.

Superheater exit steam flow sensors are shared by the Supervisory Control and Drum Level Control Systems, but median select circuits are used to prevent single sensor failures form causing an abnormal condition and resulting reactor scram. Loss of power to the median select circuits will result in a lowering of the steam drum level and a reduction in reactor power. The Plant Protection System will trip the reactor on a "low steam drum level" trip. The loss of power to the median select causes the superheater exit steam flow signal to go to zero indicating zero steam flow. This causes the steam drum level control system to close the feedwater control valves resulting in a decrease in the steam drum level. It also causes the supervisory control system to decrease reactor power in order to keep reactor power equal to plant thermal power as indicated by the superheater exit steam flow signal.

Question CS421.20

As a result of the Loose Parts Monitoring Briefing held on February 24, 1982, the staff requires a formal submittal for the following:

(1) An analysis for all loose objects that can occur in the primary, intermediate, and the steam systems and their effect on safety.

(2) An analysis for the potential effects of crud on safety.

(3) An analysis for a system to detect failures through a noise diagnostic program.

(4) Criteria for a system that will satisfy Regulatory Guide 1.33 (i.e., CRBR needs to develop their own threshold analysis).

(5) The design concepts being considered with a demonstration of feasibility.

Response

The CRBRP Project has developed and is implementing an action plan that will fully evaluate the need for CRBRP Loose Parts Monitoring. The action plan will be conducted in two phases as follows:

- Phase I - Development of the basis for CRBRP Component Degradation Monitoring
- Phase II - Identification of the requirements and implementation needed to support the basis in Phase I.

Phase I of the action plan will establish data needed to assure that the CRBRP that the CRBRP provides a level of safety comparable to LWR's and identify general approaches to obtain the needed data.

Phase II of the action plan will (a) identify the specific monitoring requirements and design changes (if any) needed to support monitoring requirements; (b) establish the operational and limiting criteria for (a); (c) determine the specific methods for implementing (a) and (b); and (d) develop a plan for research, development and test, if needed, to demonstrate the practicability of (c) above.

Phase I of the action plan is scheduled for completion by September 15, 1982 with the Phase II effort scheduled for completion by February 28, 1983. Reports on the outcome of these actions will be available after the above date.

Question CS421.23

In the CRBR PSAR Section 7.6, several instrumentation and control systems are listed as being required for safety which have not been included in the following discussion in Section 7.6. It is apparent from our review of this section that these are systems which have been omitted and also, have not been completed. The staff requires additional information to complete out review of Section 7.6.

Response

The information related to instrumentation and control for the following safety related systems is being developed and will be provided in a July 1982 amendment of PSAR Section 7.6.

1. Emergency Plant Service Water System;

2. Emergency Chilled Water System;

3. Recirculation Gas Cooling System;

4. Nuclear Island Heating, Ventilating and Air Conditioning System.

Question CS491.18 (4.3.3)

Cite references where your procedures (methods, codes, models and data) have been clearly compared with some other laboratory's procedures for the calculation of Doppler coefficients, sodium void coefficients, control rod worths, power and flux distributions, material worths, burnup, bowing reactivity coefficients, power coefficients, temperature defects, startup coefficients, etc. Some of the fundamental neutronics parameters of the CRBR are Doppler coefficients, sodium void coefficients, control rod worths, power and flux distribution, burnups, and bowing reactivity coefficients. Identify the particular safety consideration that you feel is the most impacted, limited, or made uncertain by each of the above parameters. Then, identify the most uncertain link in the calculational chain for each of the parameters.

Response

There are two major areas where published CRBRP fast reactor analysis results can be compared with independent results from other laboratories. The first is the Zero Power Plutonium Reactor (ZPPR) Cooperative Analysis Program with participation by Westinghouse, Argonne National Laboratory and General Electric. In the ZPPR analyses, cross section data and calculational methods are benchmarked against measured integral parameters (criticality, control rod worth characteristics, fission rates, sodium void worth, small-sample material worths, and others) in a zero-power full-scale mockup of the CRBRP core. The Westinghouse calculations can be compared with those from ANL and GE, as well as with the measured parameters. The results from these comparisons are used to assess bias factors and uncertainties in CRBRP nuclear performance characteristics calculated with the same methods and cross section data, a summary of which is contained in Sections 4.3.3.3 through 4.3.3.9 of the CRBRP PSAR. The following references contain details of the analysis of the ZPPR-7 experiments:

Westinghouse:	CRBRP-ARD-0237, "ZPPR-7 and 8F Cooperative Analysis Program: Critical Experiments," R. V. Rittenberger and J. A. Lake, March, 1979, (Availability: USDOE-TIC).
ANL:	Nuclear Technology 44, "Physics Studies of a Heterogeneous Liquid-Metal Fast Breeder Reactor," M. J. Lineberry, et. al., pp. 21-43, June, 1979.
GE:	CRBRP-GEFR-00025, "Analysis of the ZPPR-7 Critical Experiments," A. K. Hartman and J. T. Hitchcock, July, 1977, (Availability: USDOE-TIC).

The second area where CRBRP calculational methods can be compared with those from independent laboratories is the Large Core Code Evaluation Working Group (LCCEWG). LCCEWG is a cooperative effort, supported by USDOE, and including participants from all the major fast reactor analysis organizations* where the results of fast reactor nuclear performance characteristics, calculated with various neutronics codes, are compared. These characteristics include k_{eff}, peak-to-average fission rate, breeding ratio, control rod worth, burnup reactivity, neutron balance, and sodium void worth. The following references summarize the results of the analysis of the completed benchmark problems:

DOE/TIC-1027427, "The Large Core Code Evaluation Working Group Benchmark Analysis of a Homogeneous Fast Reactor," September, 1981.

DOE/TIC-2005709 and 2005710: "The Large Core Code Evaluation Working Group Benchmark Analysis of a Heterogeneous Fast Reactor," January, 1982.

All of the neutronic parameters listed in the question, except sodium void coefficients, factor into a variety of design basis events. They are treated at length, including uncertainties, in Chapter 15 of the PSAR. The sodium void coefficients are significant for the HCDA. This parameter, with uncertainty variations, is covered in depth in CRBRP-3, Vol. 1 (Ref. 491.18-1) and CRBRP-GEFR-00523 (Ref. 491.18-2) which are on the docket.

Ref. 491.18-1 CRBRP-3, Vol. 1, "Energetics and Structural Margin Beyond the Design Base," Dept. of Energy, CRBRP Project Office, March 1982.

Ref. 491.18-2 CRBRP-GEFR-00523, "An Assessment of HCDA Energetics in the CRBRP Heterogeneous Reactor Core," General Electric Co., Dec. 1981.

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*Participants in LCCEWG include Atomics International, Argonne National Laboratory, Combustion Engineering, General Electric, Hanford Engineering Development Laboratory, Los Alamos Scientific Laboratory, Massachusetts Institute of Technology, Oak Ridge National Laboratory, and Westinghouse Advanced Reactors Division.

Question CS490.11

The basis for construction of 99% confidence bands for the CDF fuel evaluation model was criticized by the partial draft safety evaluation report (SER) prepared in 1977 on pages 4.2-44 through 4.2-46. The question involved has not been resolved. Please discuss the relative merits of the method used in Reference (58) to Section 4.2 and the method suggested in the partial SER. Please also perform the evaluation of the two methods suggested on page 4.2-46 and provide the result. All page numbers refer to the partial draft SER.

Response

The Project is unaware that any SER for CRBRP has ever been issued by the NRC. The following response is provided on the assumption that the basis for construction of the 99% confidence bands for the CDF fuel evaluation models is a concern to the NRC.

In the CDF procedure, as outlined in Reference (58) of PSAR Section 4.2, the properties of the cladding are described by linear regression equations which were formulated using experimental data. For design level analyses, the uncertainties in the cladding properties are treated via 99% confidence bands about the regression equations.

The partial draft SER raised three issues related to the confidence bands and their mode of application in the CDF procedure. In the following response, each issue is addressed separately.

Glossary

A = intercept of regression equation,

B = slope of regression equation (i.e., regression coefficient),

N = number of data points,

S_E = standard error of estimate,

T = student's T-statistic,

V_A = variance in the intercept,

V_B = variance in the slope,

V_D = variance about the independent variable in the design environment,

$V(\hat{Y})$ = variance in the expected true value,

$\bar{X}$ = mean of independent variable from test data,

$\hat{X}$ = value of independent variable used in calculation,

$\tilde{X}$ = possible random variate about $\hat{x}$,

X_i = i-th measured value of the independent variable from test data,

$\bar{Y}$ = mean of dependent variable from test data,

$\hat{Y}$ = best estimate of the true value of the property,
$\tilde{Y}$ = possible random variate of the true value of the property,
$\tilde{Y}_D$ = possible true value of the property in the design environment,
$\tilde{Y}_m$ = possible measured value of the property in the source test environment,
Y_i = i-th measured value of the property from test data,
α = probability (confidence) level,
σ_x = standard deviation about $\bar{X}$,
σ_y = standard deviation about $\bar{Y}$,
ν = degrees of freedom.

A. Validity of the Equation Used to Compute the Confidence Bands

In the CDF technique, as described in PSAR Reference (58), the variances used to calculate the confidence bands about the regression equations are computed according to

$$V(\hat{Y}) = S_E^2 \left\{ (1/N) + (\hat{X} - \bar{X})^2 / [(N-1)\sigma_x^2] \right\} \tag{A.1}$$

The validity of the above equation has been questioned and the suggestion made by NRC that the proper variance is

$$V = S_E^2 \left\{ 1 + (1/N) + (\hat{X} - \bar{X}) / [(N-1)\sigma_x^2] \right\} \tag{A.2}$$

However, Equation (A.2) is meant to deal exclusively with predicting possible measured values of the property in the source test. Whereas, Equation (A.1) defines the variance in the true value of the property.

Clearly, the intent of the design procedure is to deal with expected values of the property in the design environment. Thus, the CDF procedure employs Equation (A.1) and adds to this the anticipated variances (i.e., uncertainties) in the independent variables as encountered in the design environment.

To highlight the fundamental differences between the two equations and their application, they are derived in parallel in the discourse which follows.

Given a test which yields N measured values of a property, Y, and the associative values the independent variable, X, then, by definition:

$$\bar{Y} = (1/N) \sum_{i=1}^{N} Y_i \quad , \tag{A.3}$$

$$\sigma_Y^2 = [1/(N-1)] \sum_{i=1}^{N} (\bar{Y} - Y_i)^2 \quad , \tag{A.4}$$

$$\bar{X} = (1/N) \sum_{i=1}^{N} X_i \quad , \tag{A.5}$$

$$\sigma_X^2 = [1/(N-1)] \sum_{i=1}^{N} (\bar{X} - X_i)^2 \quad . \tag{A.6}$$

If Y is linearly dependent on X then a linear regression equation can be formulated so that

$$\hat{Y} = A + B\hat{X} \tag{A.7}$$

describes a line (i.e., a regression line) which is a best estimate of the mean (true) value of the property when $X = \hat{X}$.

The standard error of estimate for the N measured points is given by

$$S_E^2 = [1/(N-2)] \sum_{i=1}^{N} (\hat{Y}_i - Y_i)^2 \tag{A.8}$$

where $\hat{Y}_i$ is the value of the regression equation evaluated at the test's i-th value of X. The standard error of estimate is somewhat comparable to the standard deviation vis-a-vis the regression line; it accounts for all random variations in the test data such as random measurement errors, variability in the pre-conditioning of the test specimens, variability in the test's environment and the related variability in the independent variable.

The variance in the intercept of the regression equation is given by

$$V_A = S_E^2 / N \tag{A.9}$$

and the variance in the slope is given by

$$V_B = S_\varepsilon^2 \Big/ \sum_{i=1}^{N} (\bar{X} - X_i)^2 = S_\varepsilon^2 \Big/ [(N-1)\,\sigma_X^2] \qquad \text{(A.10)}$$

The variance in $\hat{Y}$ about the regression line at $X=\hat{X}$ is obtained by summing the variances in the slope and intercept, i.e.,

$$V(\hat{Y}) = V_A + V_B(\hat{X} - \bar{X})^2 \qquad \text{(A.11)}$$

Note that $V(\hat{Y})$ is the variance in the estimate of the property's true value as determined from the test and is not the variance in the test data.

Assume for a moment that all collateral variabilities need not be considered. In this case, a single possible value of the true property at $X=\hat{X}$ would be given by

$$\tilde{Y} = \hat{Y} + RV(\hat{Y})^{1/2} \qquad \text{(A.12)}$$

where R is understood to be a suitable multiplicative factor. In other words, Equation (A.12) defines the variability of the expected true value of the property if X were known precisely and if there were no extrinsic random errors.

In reality, the overall variability in the expected value of the property depends on the collateral variabilities that are operating under any given set of conditions; these collateral variances must, of course, be added to Equation (A.12).

If one were interested in estimating a possible measured value of the property in the source test* itself then the appropriate collateral variance must include the random measurement errors, the variability in pre-conditioning, etc. This, of course, is described directly by the standard error of estimate given by Equation (A.8). It follows therefore, that in the source test, a possible measured value of the property at $X=\hat{X}$ is given by

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*A source test is defined as the test used to obtain the N measured data points one which is identical in every way.

$$\tilde{Y}_m = \hat{Y} + RV(\hat{Y})^{1/2} + RS_E \qquad \text{(A.13)}$$

Expanding Equation (A.13) yields

$$\tilde{Y}_m = A + B\hat{X} + RS_E\left\{1 + (1/N) + (\hat{x}-\bar{x})^2/[(N-1)\,\sigma_x^2]\right\}^{1/2} \qquad \text{(A.14)}$$

Note that the variance employed in Equation (A.14) is exactly the form suggested by NRC [i.e., see Equation (A.1)] and that (A.14) is relevant only in computing possible measured values in the source test.

If the property in question is to be used in a design calculation, then there is no interest in the collateral variance associated with the source test's random errors. Indeed, for a design calculation, the appropriate collateral variance(s) must reflect the uncertainties in the design environment as well as the specific nature of the design problem.

Typically, the collateral variances associated with the design environment are treated via an assigned uncertainty in the independent variable. Thus, let V_D be the assigned variance about $\hat{x}$ in the design environment. Then, a possible true value of the property in the design environment is given by

$$\tilde{Y}_D = \hat{Y} + R_1 V(\hat{Y})^{1/2} + B R_2 V_D^{1/2} \qquad \text{(A.15)}$$

Expanding Equation (A.15) yields

$$\tilde{Y}_D = A + B\left[\hat{x} + R_2 V_D^{1/2}\right] + R_1 S_E\left\{(1/N) + (\hat{x} + R_2 V_D^{1/2} - \bar{x})^2/[(N-1)\,\sigma_x^2]\right\}^{1/2} \qquad \text{(A.16)}$$

Let $\tilde{x}$ be a possible random variate of the independent parameter, i.e.,

$$\tilde{x} = \hat{x} + R_2 V_D^{1/2} \qquad \text{(A.17)}$$

Then, it follows from Equation (A.16) that

$$\tilde{Y}_D = A + B\tilde{x} + R_1 S_E\left\{(1/N) + (\tilde{x}-\bar{x})^2/[(N-1)\,\sigma_x^2]\right\}^{1/2} \qquad \text{(A.18)}$$

Equation (A.18) is a direct expression of the typical application of a regression equation to a design problem. Specifically, the design level estimate of the true property is computed using the regression equation and its variance, both evaluatd at the design level value for the independent variable.

Equation (A.18) represents the procedure employed by the CDF technique to compute cladding properties; note that the variance used in (A.18) is exactly that specified in PSAR Reference (58).

B. The Distribution of Data Points Relative to the Confidence Points About a Regression Equation.

In Section IV of PSAR Reference (58), a series of Figures compare the regression equations to their source data, included in these figures are the 99% confidence bands about the regression line. In all cases, a large fraction of the data fall outside the confidence bands.

The question has been frequently asked as to why 99% of the data are not enclosed with the confidence bands and condequently, questions the bands' validity.

As conventionally defined, confidence limits address the probability that the true value of the property (i.e., the mean) lies within a certain range. Thus, there is no requirement that the confidence limits encompass the data.

On the other hand, the limits which are expected to encompass a specified fraction of the data are frequently referred to as tolerance limits; these limits may be used to address the probability that a measured value (i.e., a data point) will fall within a certain range.

Conventionally defined confidence bands about a regression equation can be described by Equation (A.18) with $\tilde{X} = \hat{X}$ and $|R_1|$ taken to be a proper value of the student's T-statistic. In other words,

$$\tilde{Y} = A + B\hat{X} \pm T S_E \left\{ (1/N) + (\hat{X} - \bar{X})^2 / [(N-1)\, \sigma_X^2] \right\}^{1/2} \qquad (B.1)$$

Similarly, Equation (A.14) would describe the tolerance bands, i.e.,

$$\tilde{Y}_m = A + B\hat{X} \pm TS_E \left\{ 1 + (1/N) + (\hat{X} - \bar{X})^2 / [(N-1)\sigma_X^2] \right\}^{1/2} \quad \text{(B.2)}$$

Equation (B.1) addressed the probability that the true value of the property lies with the range $\hat{Y} \pm |\tilde{Y} - \hat{Y}|$; Equation (B.2) addresses the probability that data points will fall with the range $\hat{Y} \pm |\tilde{Y}_m - \hat{Y}|$. Clearly $|\tilde{Y} - \hat{Y}| < |Y_m - \hat{Y}|$; thus, one would anticipate that data points may fall outside the confidence bands.

C. Conservativeness of the CDF Design Procedure.

As mentioned earlier, for design level analyses, the uncertainties in the cladding's properties are treated in the 99% confidence bands about their regression equations; these are taken to combine in the most unfavorable way possible. Depending on the property in question the most unfavorable level may be the upper or lower confidence band.

In Section V of PSAR Reference (58), the design procedure is examined and verified using a number of independent sources of data. In these verification examples, the design procedure successfully bracketed, or would have precluded, all of the failure data from (a) multistage stress-rupture tests, (b) creep/tensile and creep/burst tests and (c) FCTT tests.

The veracity of the above verifications has been questioned*. Specifically, it has been suggested that if the confidence bands do not encompass all of the data from the respective source tests, then the limits computed in the verification examples could not have bracketed the failures as shown.

In view of the discussions in Parts A and B, it should now be clear that, within the capabilities of the models, the design procedure should bracket a vast majority of the failures. Specifically, it was shown that the confidence bands describe the limits of certainly about the true values of the properties.

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*All of the information and data necessary for the duplication of the verification analyses are contained within PSAR Reference (58).

Thus, the design level CDF, computed with the worst combination of 99% confidence bands should achieve unity prior to any CDF computed with random combinations of the properties as might be encountered in nature.

It was with this point in mind that the following study was undertaken.

The objective of the study was to compare the results of a typical design level fuel pin analysis with that from a Monte Carlo analysis. In the Monte Carlo treatment, the values for the various properties are randomly chosen from simulated experimental populations and then combined at random. This is done in a large number of FURFAN computations. Thus, within the capabilities of the models, the Monte Carlo treatment yields a sample of CDF values drawn from a population having variability comparable to that which would be obtained in nature. In this regard, the Monte Carlo distributions in the CDF may be taken as estimates of the conditional probability of achieving given CDF values.

The operating parameters assumed for this analysis are summarized in Table QCS490.11-1. These parameters correspond to conditions with 2σ plant factors prevailing; note that these parameters remain the same within each Monte Carlo trial.

The results of the typical deterministic design level analysis for the subject pin is given in Figure QCS490.11-1. As shown, by combining all properties at their worst confidence levels a CDF = 1.0 is achieved after 540 EFPD. In comparison, a combination of properties at their nominal levels (with the 2σ factors still prevailing) yields a CDF of 0.905 after 825 EFPD.

The Monte Carlo analysis involved 100 trials each using a set of properties which were randomly chosen from their respective distributions and randomly combined. Figure QCS490.11-2 gives the resultant distributions of CDF values at 540 EFPD, the time the design limit is achieved. Note that at this time, the entire population of possible CDF values is below unity with 99% of the values less than 0.8.

Figure QCS490.11-3 shows the cumulative distributions in the CDF's at various times during trhe 825 EFPD operating period. As such, this figure represents

estimates of the time dependent, cumulative conditional probabilities of achieving CDF values less than some specific value, i.e., P(CDF $\leq$ X).

Figure QCS490.11-4 compares the culuiative conditional probability estimate for CDF $\leq$ 1.0 to the design level and nominal CDF's. As shown, at the 540 EFPD limit P(CDF $\leq$ 1) ~ 100%. At 825 EFDP, where the nominal CDF is 0.91, P(CDF $\leq$ 1) ~ 54%. In other words, at the steady state limit (i.e., 540 EFPD) there is almost a 100% probability that the CDF is less than unity; also, after 825 EFPD there is ~ 54% probability that CDF $\leq$ 1.0.

TABLE QCS490.11-1 SUMMARY OF PARAMETERS USED IN ANALYSIS

Inside Radius of Cladding: 0.100 Inch
Outside Radius of Cladding: 0.114 Inch
Axial Location: X/L = 1.0
2σ Plant and 2σ Hot Spot Factors

TIME	TEMPERATURE (^{o}F)		PRESSURE
(DAYS)	I.D.	O.D.	(PSI)
0	1308	1284	180
275	1236	1211	500
275	1259	1234	540
559	1195	1170	1150
550	1214	1187	1150
825	1152	1128	1750

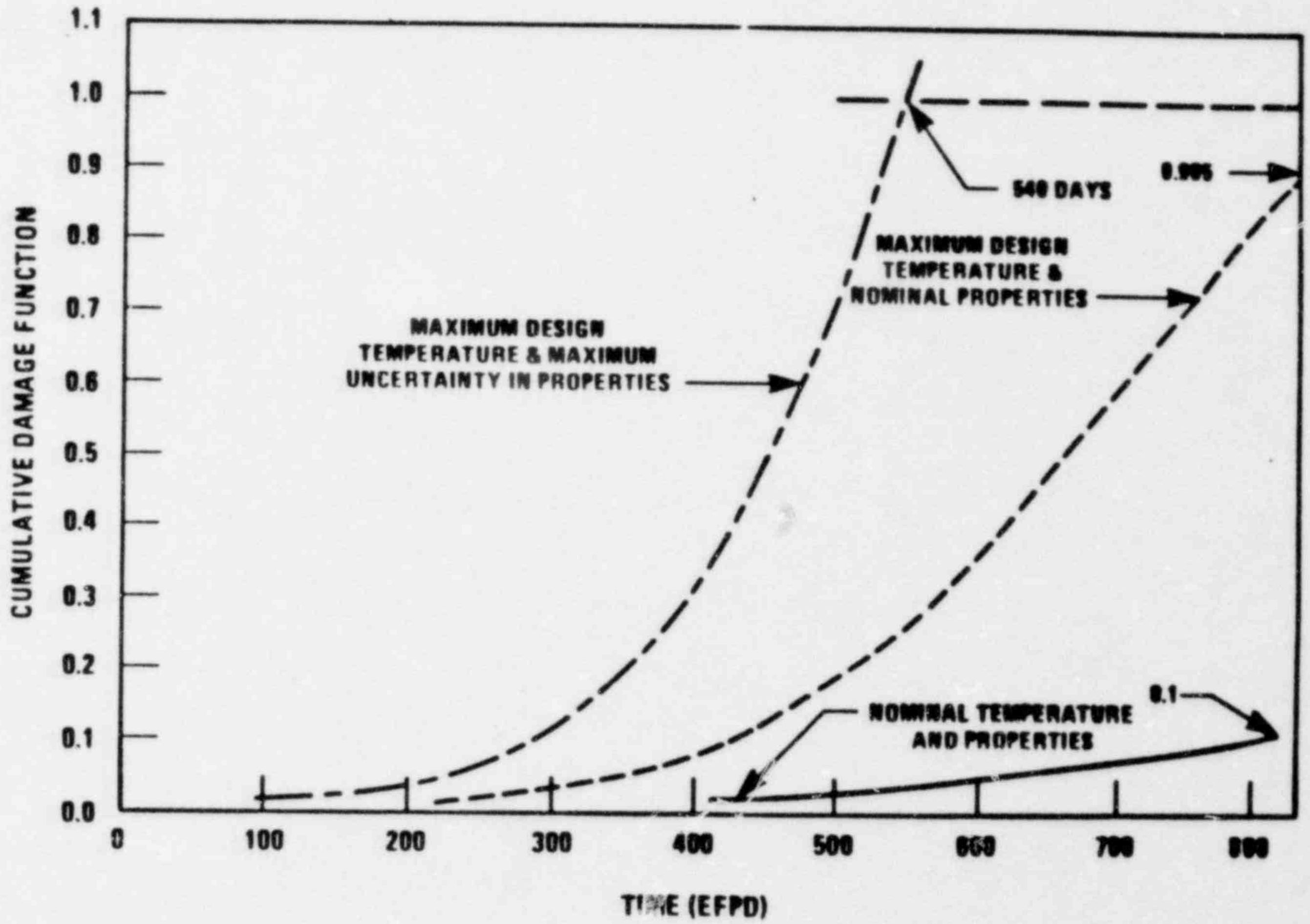

Figure QCS490.11-1. Cumulative Mechanical Damage Due to Steady State Operation at X/L = 1.0

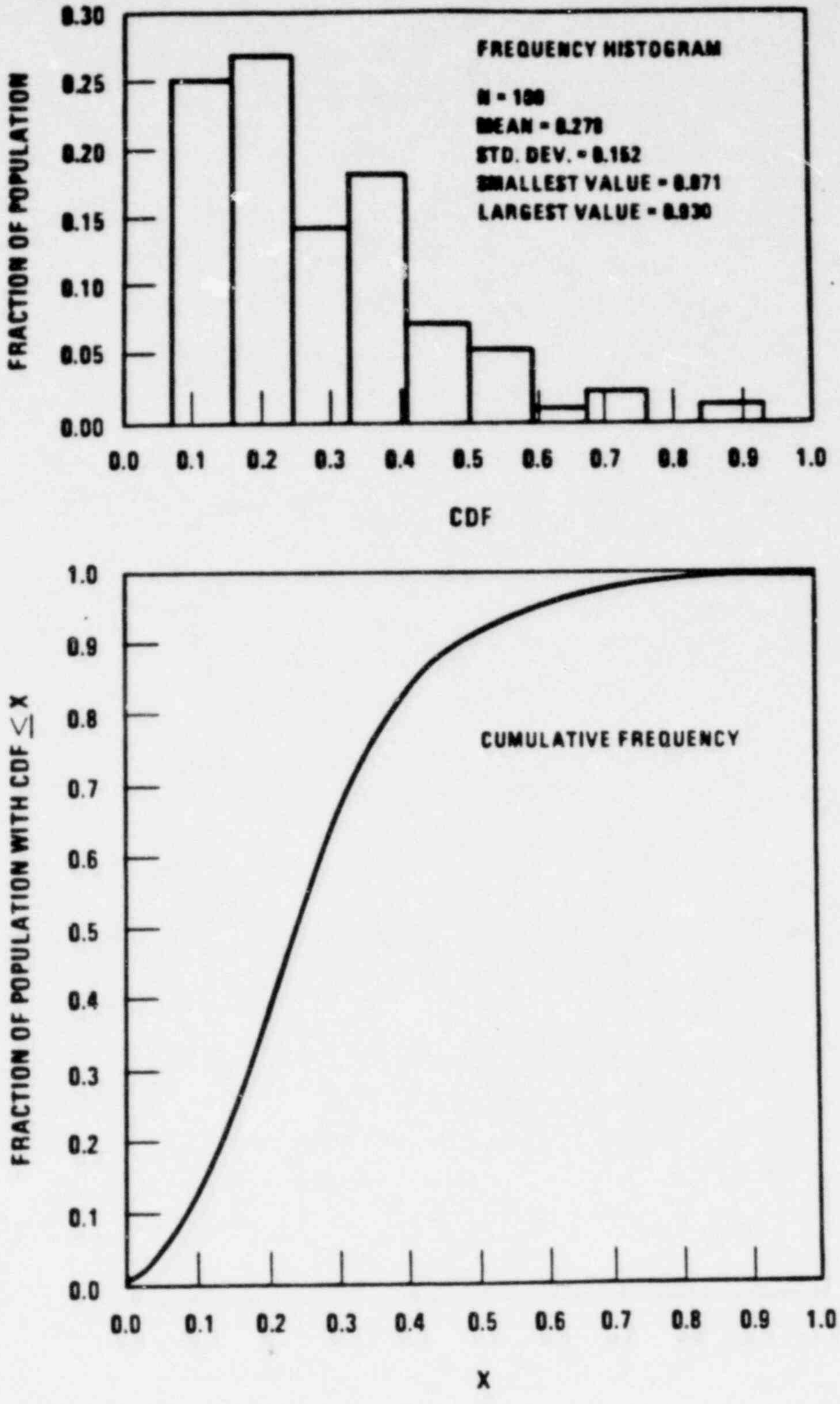

Figure QCS490.11-2. Distribution of Possible CDF Values at X/L = 1.0 After 540 EFPD of Steady State Operation

7190-2

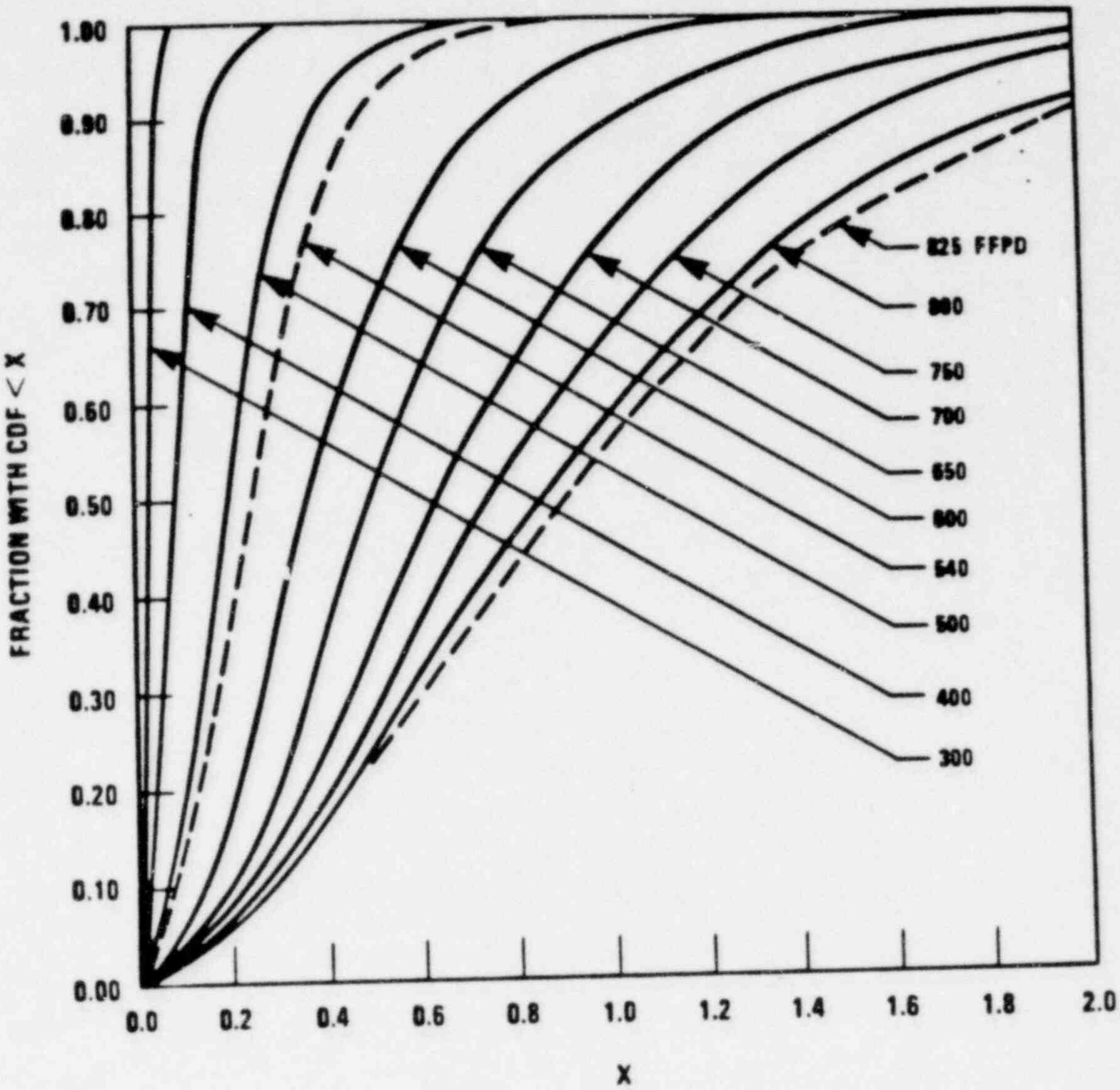

Figure QCS490.11-3. Cumulative Frequency Distributions in Possible CDF Values at X/L = 1.0
After Various Periods of Steady State Operation

7190-3

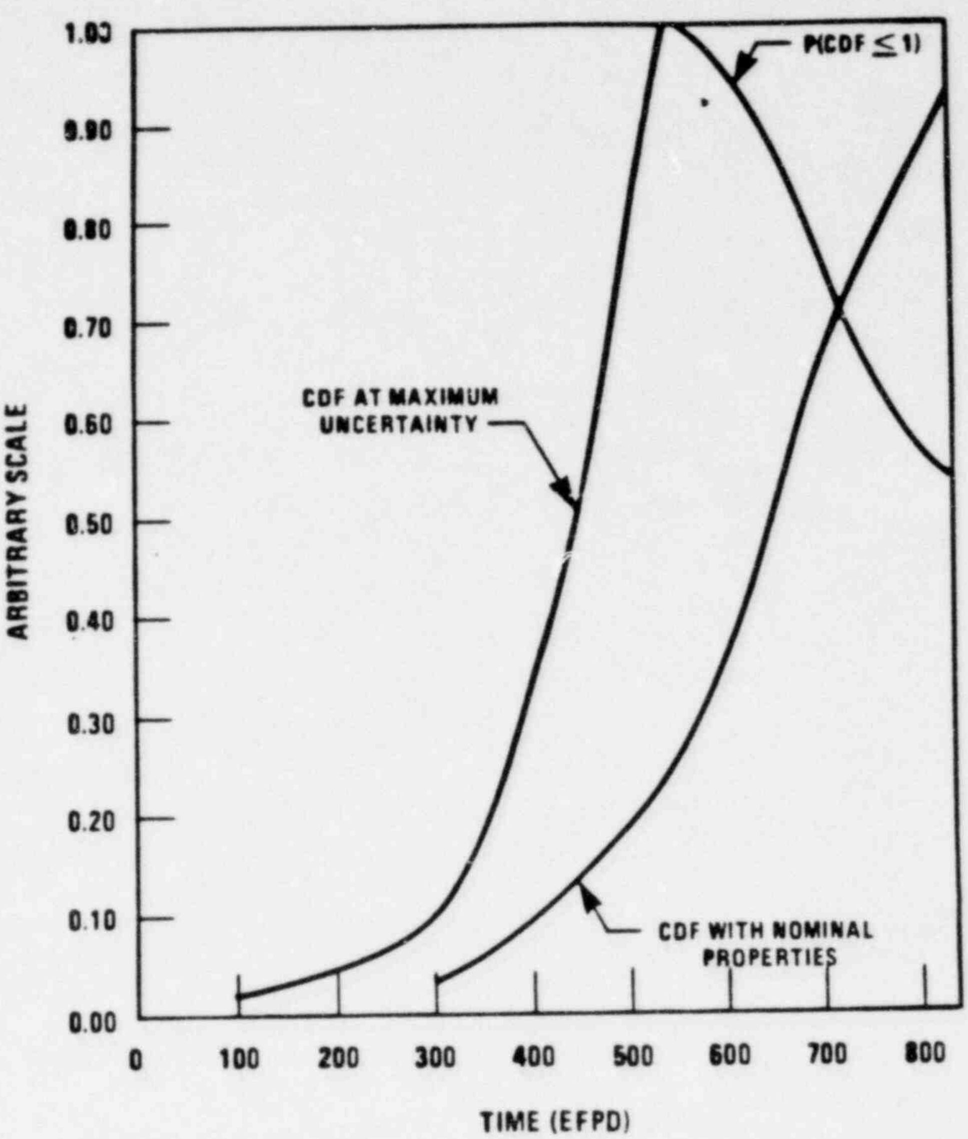

Figure QCS490.11-4. Comparison of the Conditional Probability that CDF ≤ 1.0 and the Corresponding CDF Values at the Design and Nominal Levels

7190-4

Question CS490.35

Many computer codes were used by the CRBRP designers to perform the thermal and hydraulic analyses presented in section 4.4 of the CRBR PSAR. Some of these codes are proprietary and some were developed by the CRBRP or its contractors and are not widely available. To evaluate the applicability of these codes to the thermal and hydraulic analyses presented in section 4.4, substantially more information is needed than is presented in section 4.4, in Appendix A, and in the references cited in Appendix A. Therefore, please provide code manuals and/or detailed descriptions along with code listings for the following codes.

a. CATFISH
b. CORINTH
c. COTEC
d. CRSSA
e. DEMO
f. FATHOM-360
g. FATHOM-360S
h. FLODISC
i. FORE-2M
j. NICER
k. OCTOPUS
l. TRITON

Response

Since last submission of the CRBRP PSAR, two of the codes in the above list have been superseded. The CØRINTH code has been replaced by the DOE national program code CØBRA-WC. Also, the FLØDISC code has been replaced by CØBRA-WC for analysis of very low flow (natural circulation) conditions. In the flow range from 100% to ~ 10% flow the FLØDISC code has been replaced by CATFISH.

The CRAB-II code which analyzes the primary control assembly steady state hydraulics, scram dynamics and flotation behavior should be added to the list.

Descriptions of the major features, models applications and typical results of the above codes have been reported in the open literature; a list of these papers/articles/reports is attached.

An extensive validation effort is ongoing for all of the above codes.

Manuals and validation reports for all the above codes will be provided prior to FSAR submittal. Appendix A will be modified to reflect the above changes.

AVAILABLE OPEN LITERATURE PUBLICATIONS

CATFISH

1) M.D. Carelli and J.M. Willis, "An Analytical Method to Accurately Predict LMFBR Core Flow Distribution", Trans. Amer. Nucl. Soc., 32, pp. 575-576, 1979.

2) M.D. Carelli and J.M. Willis, "Analytical Modeling of Core Hydraulics and Flow Management in Breeder Reactors", Proceedings of the XVIII Congress of the International Association for Hydraulic Research, Cagliari (Italy), September 10-14, 1979.

CØTEC

1) E.H. Novendstern, "Mixing Model for Wire Wrap Fuel Assemblies", Trans. Amer. Nucl. Soc., 15, pp. 866-867, 1972.

2) E.H. Novendstern, "Turbulent Flow Pressure Drop Model for Fuel Rod Assemblies Utilizing a Helical Wire Wrap Spacer system", Nucl. eng. Design, 22, pp. 19-27, 1972.

3) Y.S. Tang, M.R. Yeung and M.D. Carelli, "A Core Design Subchannel Ana'ysis Code Calibration and Validation", to be presented at the ANS Annual Meeting, Los Angeles, June 1982.

4) F.C. Engel, R. A. Markley and B. Minushkin, "Buoyancy Effects on Sodium Coolant Temperature Profiles Measured in an Electrically Heated Mockup of a 61-Rod Breeder Reacter Blanket Assembly", ASME-78-WA-HT-25.

5) F.C. Engel, R.A. Markley and B. Minushkin, "Heat Transfer Test Data of a 61-Rod Electrically Heated LMFBR Blanket Assembly Mockup and Their Use for Subchannel Code Calibration", in Fluid flow and Heat Transfer Over Rod or Tube Bundles, pp. 223-229, American Society of Mechanical Engineers, New York, 1979.

6) F.C. Engel, R.A. Markley and B. Minushkin, "The Effect of Heat Input Patterns on Temperature Distribution in LMFBR Blanket Assemblies", ANS/ASME International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, Saratoga Springs, NY, October 1980, NUREG/CP-0014, Vol. 3.

DEMØ

1) W.H. Allston, "LMFBR Demonstraton Plant Simulation Model (DEMØ)", CRBRP-ARD-0005, February 1978.

FATHØM 360 AND FATHØM-360S

1) M.D. Chuang, M.D. Carelli, C.W. Bach and J.S. Killimayer, "Three-Dimensional Thermal-Hydraulic Analysis of Wire Wrapped Rods in Liguid Metal Fast Breeder Reactor Core Assemblies", Nuclear Science and Eng., 64, pp. 244-257, 1977.

2) M.C. Chuang, M.D. Carelli and M.R. Yeung, "Distributed Parameter Analyses of the Thermal-Hydraulic Behavior of Wire Wrapped Rods in LMFBR Cores", paper submitted to the 2nd International Topical Meeting on Nuclear Reactor Thermalhydraulics, Santa Barbara, January 1983.

NICER

1) M.D. Carelli, C. W. Bach and R.A. Markley, "Analytical Techniques for Thermalhydraulics Design of LMFBR Assemblies", Trans. Amer. Nucl. Soc., 17, pp. 423-424, 1973.

ØCTØPUS

1) M.D. Carelli, A.J. Friendland, C.W. Bach and R.A. Markley, "An Optimized Method for Orificing LMFBR Cores", Trans. Amer. Nucl. Soc., 26, pp. 437-438, 1977.

2) M.D. Carelli and C.W. Bach, "Orificing Interchangeable LMFBR Cores", Trans. Amer. Nucl. Soc., 34, pp. 268-270, 1980.

TRITØN

1) M.D. Carelli and C.W. Bach, "Thermal-Hydraulic Analyses for CRBRP Core Restraint Design", Trans. Amer. Nucl. Soc., 21, pp. 393-395, 1975.

2) F.C. Engel, B. Minushkin and R.A. Markley, "Comparisons of Design Code Predictions with LMFBR Blanket Heat Transfer Test Results", American Nuclear Society and the European Nuclear Society November 1980 International Conference, Washington, D.C.

CRAB and CRAB-II

1) M.D. Carelli, C.W. Bach and R.A. Markley, Hydraulic and Scram Dynamics Analysis of LMFBR Control Rod Assemblies", Trans. Amer. Nucl. Soc., 16, 1, pp. 218-219, 1973.

2) M.D. Carelli, H.W. Brandt, C.W. Bach and H.D. Kulikowski, "LMFBR Control Rods Scram Dynamics", Trans. Amer. Nucl. Soc., 18, pp. 278-279, 1974.

3) M.D. Carelli, L.A. Baker, J.M. Willis, F.C. Engel and D.Y. Nee, "CRAB-II: A Computer Program to Predict Hydraulics and Scram Dynamics of LMFBR Control Assemblies and Its Validation", to be presented at the ANS Topical meeting on Reactor Physics and Core Thermal-Hydraulics, Kiamesha Lake, NY, September 1982.

CØBRA-WC

1) T.L. George, K.L. Basehore, C.L. Wheeler, W.A. Prather and R.E. Masterson, "CØBRA-WC: A Version of CØBRA for Simple-Phase Multi-Assembly Thermal-Hydraulic Transient Analysis", PNL-3259, July 1980.

2) E.U. Khan, W.A. Prather, T.L. George, J.M. Bates, "A Validation Study of the CØBRA-WC Computer Program for LMFBR Core Thermal-Hydraulic Analysis", PNL-4128, December 1981.

FØRE-2M

1) J.N. Fox, B.E. Lawler and H. R. Butz, "FØRE-II: A Computational Program for the Analysis of Steady State and Transient Reactor Performance", GEAP-5273, September 1966.

2) J.V. Miller and R.D. Coffield, "FØRE-2M: Modified Version of the FØRE-II Computer Program for the Analysis of LMFBR Transients", CRBRP-ARD-0142 (available from US/DOE Technical Information Center), November 1976.