STATE OF PENNSYLVANIA

COUNTY OF ALLEGHENY

T. M. Anderson, being duly sworn, states that he is Manager, Nuclear Safety Department, Water Reactor Divisions of the Westinghouse Electric Corporation; that he is authorized on the part of said Corporation to sign and file with the Nuclear Regulatory Commission this application and exhibits attached thereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information and belief.

T. M. Anderson

T. M. Anderson

Subscribed and sworn to me, a Notary Public in and for State and County above named, this 5 day of December, 1978.

Rebecca L. Beyer

7901160164

RESAR-41
AMENDMENT 24
INSTRUCTION SHEET

The following instructional information and check list is being furnished to insert Amendment 24 into the RESAR-41 Reference Safety Analysis Report.

Since in most cases the original RESAR-41 contains information printed on both sides of a sheet of paper, a new sheet is furnished to replace sheets containing superseded material. As a result, the front or back of a sheet may contain information that is merely reprinted rather than changed.

Discard the old sheets and insert the new sheets, as listed below. Keep these instruction sheets in the front of Volume I to serve as a record of changes.

Remove (Front/Back)	Insert (Front/Back)
3-vii/3-viii	3-vii/3-viii
3-ix/3-x	3-ix/3-x
3-xi	3-xi
	Appendix 3B should be inserted in RESAR-41 immediately following Appendix 3A in Volume 1.

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LIST OF FIGURES

24

APPENDIX 3B

EXTENSION REVIEW MATTERS FOR PRELIMINARY DESIGN APPROVALS

The Commission's August 22, 1978, policy statement on standardization includes a provision which allows any Preliminary Design Approval (PDA) that had been previously issued for a three-year term to be extended for two additional years. This provision applies to PDA-3 issued for the RESAR-41 application on December 31, 1975, and when approved, will extend the PDA to December 31, 1980.

As set forth in the policy statement, each application for a PDA extension will be subject to an assessment of the design with respect to the Category I, II, III and matters approved since the regulatory requirement's cutoff date for the PDA in question and the Category IV matters scheduled for review by the R^3C. A tabulation of each Category I, II, III, and IV matter approved or scheduled for review since the regulatory requirement's cutoff date of October 23, 1974 for RESAR-41, is provided in the index tables which follow.

Appendix 3B addresses each matter identified by the index as being within Westinghouse scope of supply. Matters not within Westinghouse scope will be addressed in the applicant's Safety Analysis Report.

As noted in Appendix 3A, throughout the text of RESAR-41, commitments are made to comply with regulatory criteria, positions and guides. Compliance is based upon the Westinghouse interpretation of the statement of the requirement. Where the Westinghouse position differs from Westinghouse's understanding of the regulatory requirement, alternate positions are presented and defended as acceptable.

The regulatory requirement's cutoff date for RESAR-41 is October 23, 1974, which precedes the effective dates and implementation dates of all of the PDA extension review matters in Categories I, II, III and IV. Therefore, according to present NRC policy on standardization, RESAR-41 compliance is not required for any of the Category I matters. However, as indicated in the individual responses which follow, compliance or an acceptable alternative position can be demonstrated for all of the Westinghouse NSSS scope PDA extension review matters in Categories I, II, III and IV, without the necessity for implementing RESAR-41 design changes.

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CATEGORY I MATTERS APPROVED BY REGULATORY REQUIREMENTS

REVIEW COMMITTEE (R^3C)

Item	Reg. Guide	Rev.	Effective Date	W NSSS Scope	Applicant Scope
1	1.7	2	1/31/78	X	
2	1.9	1	9/1/78		X
3	1.20	2	1/9/76	X	
4	1.28	1	11/29/77	X	
5	1.29	3	6/20/78	X	
6	1.31	2	7/20/76	X	
7	1.32	2	11/14/77		X
8	1.33	1	10/21/76		X
9	1.35	2	8/15/75		X
10	1.38	2	5/77	X	
11	1.39	2	7/12/77		X
12	1.52	2	11/29/77		X
13	1.63	1	3/22/77		X
14	1.64	2	1/9/76	X	
15	1.68	2	6/20/78	X	X
16	1.68.1	0	9/26/75	Not Applicable to PWRs	
17	1.72	1	11/15/77		X
18	1.84	12	3/78	X	
19	1.85	12	3/78	X	
20	1.90	1	5/26/77		X
21	1.92	1	8/22/75	X	
22	1.94	1	2/6/76		X
23	1.95	1	10/21/76		X
24	1.99	1	1/14/77	X	See Cat. III Response
25	1.100	1	6/14/77	X	
26	1.103	1	10/76		X
27	1.106	1	1/28/77		X
28	1.107	1	10/21/76		X
29	1.116	0-R	5/77		X
30	1.118	1	9/27/77	X	

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CATEGORY I MATTERS (Continued)

Item	Reg. Guide	Rev.	Effective Date	W NSSS Scope	Applicant Scope
31	1.120	1	5/11/77		X
32	1.122	1	11/15/77		X
33	1.123	1	7/77	X	
34	1.126	0	1/14/77	X	
35	1.128	1	6/20/78		X
36	1.129	0	2/18/77		X
37	1.131	0	5/26/77		X
38	1.132	0	5/11/77		X
39	1.134	0	3/22/77		X
40	1.135	0	7/12/77		X
41	1.136	0	8/31/77		X
42	1.137	0	9/27/77		X
43	NUREG-0102 (SRP 1.8)	0	9/27/77	X	
44	1.138	0	11/15/77		X
45	1.XXX	0	11/15/77		X
46	1.140	0	11/29/77		X
47	1.142	0	1/31/78		X
48	8.19	0	3/14/78	X	
49	RSB 5-2	0	3/14/78	X	See Cat. III Response

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CATEGORY II MATTERS APPROVED BY REGULATORY REQUIREMENTS REVIEW COMMITTEE (R^3C)

Item	Reg. Guide	Rev.	Effective Date	W NSSS Scope	Applicant Scope
1	1.27	2	11/12/75		X
2	1.52	1	1/9/76		X
3	1.59	2	8/77		X
4	1.63	1	3/22/77		X
5	1.68.2	1	5/16/78		X
6	1.91	1	11/15/77		X
7	1.97	1	1/28/77	X	
8	1.102	1	11/12/75		X
9	1.105	1	9/15/76	X	
10	1.108	1	6/14/77		X
11	1.115	1	3/22/77		X
12	1.117	1	12/20/77		X
13	1.124	1	8/31/77	X	
14	1.130	0	7/77	X	
15	1.137	0	9/29/77		X
16	8.8	2	8/18/76	X	
17	BTPASB 9.5-1		8/18/76		X
18	BTP MTEB 5-7		4/13/77		X
19	SRP 5.4.7	1	1/31/78	X	See Cat. III Response
20	1.141	0	1/31/78	X	See Cat. III Response

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CATEGORY III MATTERS APPROVED BY REGULATORY REQUIREMENTS REVIEW COMMITTEE (R^3C)

Item	Reg. Guide	Rev.	Effective Date	W NSSS Scope	Applicant Scope
1	1.56	1	5/16/78	Not Applicable to PWRs	
2	1.68.2	1	5/16/78		X
3	1.99	1	1/14/77	X	
4	1.101	1	3/77		X
5	1.114	1	11/76		X
6	1.121	0	5/11/76	X	
7	1.127	1	11/29/77		X
8	1.137	0	9/27/77		X
9	SRP 5.4.7	1	1/31/78	X	
10	1.141	0	1/31/78	X	
11	RSB 5-2	0	3/14/78	X	

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CATEGORY IV MATTERS

A. REGULATORY GUIDES NOT CATEGORIZED

Item	Reg. Guide	Rev.	Issue Date	W NSSS Scope	Applicant Scope
1	1.13	1	12/75		X
2	1.14	1	8/75	X	
3	1.75	1	1/75	X	
4	1.79	1	9/75	X	
5	1.83	1	7/75	X	
6	1.89	0	11/74	X	
7	1.93	0	12/74		X
8	1.104	0	2/76		X

B. SRP CRITERIA

Item	Branch	SRP Criteria	Implementation Date	W NSSS Scope	Applicant Scope
1	MTEB	5.4.2.1	11/24/75	X	
2	CSB	6.2.1 6.2.1A 6.2.1B 6.2.1.2 6.2.1.3 6.2.1.4 6.2.1.5	11/24/75	X	
3	CSB	6.2.5	11/24/75	X	
4	CSB	6.2.3	11/24/75		X
5	CSB	6.2.4	11/24/75	X	
6	ASB	9.1.4	11/24/75	X	
7	ASB	10.4.9	11/24/75		X
8	SEB	3.5.3	11/24/75	X	

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CATEGORY IV MATTERS

B. SRP CRITERIA (Continued)

Item	Branch	SRP Criteria	Implementation Date	W NSSS Scope	Applicant Scope
9	SEB	3.7.1	11/24/75		X
10	SEB	3.7.2	11/24/75	X	
11	SEB	3.7.3	11/24/75	X	
12	SEB	3.8.1	11/24/75		X
13	SEB	3.8.2	11/24/75		X
14	SEB	3.8.3	11/24/75	X	
15	SEB	3.8.4	11/24/75		X
16	SEB	3.8.5	11/24/75		X
17	SEB	3.7	11/24/75	X	
		11.2	11/24/75		
		11.3	11/24/75		
		11.4	11/24/75		
18	SEB	3.3.2	11/24/75		X
19	SEB	3.4.2	11/24/75		X
20	ASB	10.4.7	11/24/75	X	
21	AB	4.4	11/24/75	X	
22	RSB	5.2.5	11/24/75	X	
23	RSB	3.2.2	11/24/75		X

C. OTHER POSITIONS

Item	Branch	SRP Criteria	Implementation Date	W NSSS Scope	Applicant Scope
1	SEB	3.5.3	12/1/76		X
2	SEB	3.7.1	8/1/76	X	
3	SEB	3.8.1	4/1/76		X
		3.8.2			
4	SEB	3.8.4	9/1/76		X
5	SEB	3.5.3	10/1/76		X
6	RSB	6.3	6/1/77	X	

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CATEGORY IV MATTERS

C. OTHER POSITIONS (Continued)

Item	Branch	SRP Criteria	Implementation Date	W NSSS Scope	Applicant Scope
7	RSB	6.3	9/1/77	X	
8	RSB	15.1.5	4/1/77	X	
9	RSB	5.4.6 5.4.7 6.3	12/1/77	X	
10	RSB	3.5.1	3/28/78	X	
11	AB	4.4	1/1/77	X	
12	PSB	8.3	1/1/78		X
13	CSB	6.2.1.2	6/1/76	X	
14	CSB	6.2.6	9/1/77		X
15	CSB	6.2.1.4	1/1/77	X	
16	ASB	3.6.1 3.6.2	11/1/77		X
17	ASB	9.2.2	1/1/77	X	
18	ASB	10.4.7	8/1/76	X	
19	ICSB	3.11	1/1/76		X

Category I

Item

RG 1.7 Rev. 2 1/31/78	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident

RESPONSE

Calculations for post LOCA hydrogen analysis in RESAR-41 Section 15.4.1.2 are in compliance with Regulatory Guide 1.7 Rev. 1. The hydrogen production rates resulting from corrosion and radiolysis are calculated according to methods and utilizing parameters, given in the guide. Compliance with RG 1.7 Rev. 2 will be calculated when the revised guide is issued. However, because the Regulatory Requirement's cutoff date for RESAR-41 precedes 1/31/78, compliance with Rev. 1 of the guide is acceptable.

Category I

Item

RG 1.20 Rev. 2 1/9/76	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing

RESPONSE

The extent to which the requirements in Regulatory Guide 1.20 Rev. 2 are satisfied for Reactor Internals is discussed in Section 3.9 of RESAR-41.

Category I

item

RG 1.28 Rev. 1 11/29/77	Quality Assurance Program Requirements (Design and Construction)

RESPONSE

The Westinghouse Quality Assurance Programs, as described in WCAP-8370 Rev. 8A[1] and WCAP-7800 Rev. 5[2], satisfy the requirements of Regulatory Guide 1.28 Rev. 0, which contains requirements similar to those in Regulatory Guide 1.28 Rev. 1.

Category I

Item

RG 1.29 Rev. 3 6/20/78 — Seismic Design Classification

RESPONSE

The position described in Appendix 3A to RESAR 41 for Regulatory Guide 1.29 Rev. 1 is also applicable for Regulatory Guide 1.29 Rev. 3.

Category I

Item

RG 1.31 Rev. 2 — Control of Ferrite Content in Stainless Steel Weld Metal

RESPONSE

Regulatory Guide 1.31, Revision 2, Control of Ferrite Content in Stainless Steel Weld Metal, describes a method for implementing General Design Criteria 1 of Appendix A 10CFR Part 50 and Appendix B 10CFR Part 50 with regard to control of welding austenitic stainless steel components and systems. The following paragraphs discuss the method to be used by Westinghouse to control delta ferrite in austenitic stainless steel welding; this method is in compliance with Revision 2 of the guide.

The welding of austenitic stainless steel is controlled to mitigate the occurrence of microfissuring or hot cracking in the weld. Although published data and experience have not confirmed that fissuring is detrimental to the quality of the weld, it is recognized that such fissuring is undesirable in a general sense. Also, it has been well documented in the technical literature that the presence of delta ferrite is one of the mechanisms for reducing the susceptibility of stainless steel welds to hot cracking. Available data indicates that a minimum delta ferrite level (expressed in Ferrite Number (FN)), above which the weld metals commonly used by Westinghouse will not be prone to hot cracking, lies somewhere between 0 FN and 3 FN.

The scope of these controls discussed herein encompasses welding processes used to join stainless steel parts in components designed, fabricated or stamped in accordance with ASME B&PV Code, Section III Class 1, 2, and CS components. Delta ferrite control is appropriate for the above welding reqirements except where no filler metal is used or for other reasons such control is not applicable. These exceptions include electron beam welding, autogenous gas shielded tungsten arc welding, explosive welding, and welding using fully austenitic welding materials.

Westinghouse components are fabricated utilizing welding procedures qualified in accordance with ASME Code Section III; also delta ferrite content verification is required for welding materials used for welding qualification testing and for each welding process used in the production of austenitic stainless steel components. Specifically, undiluted weld deposits of the "starting" welding materials are required to contain a minimum delta ferrite level of 5 FN. The ASME Code permits the use of either a chemical analysis method or a magnetic measurement method to determine the delta ferrite content; however, in the fabrication of Westinghouse components, Ferrite Number is measured on an as-deposited weld pad using a calibrated magnetic measuring device, as recommended by Revision 2 of Regulatory Guide 1.31.

Category I

Item

RG 1.38 Rev. 2 5/77	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants

RESPONSE

The Westinghouse position on compliance with Regulatory Guide 1.38, Rev. 2 is presented in WCAP-8370 Rev. 8A[1] and WCAP-7800 Rev. 5[2].

Category I

Item

	RG 1.64 Rev. 2 1/9/76	Quality Assurance Requirements for the Design of Nuclear Power Plants

RESPONSE

The Westinghouse position on compliance with Regulatory Guide 1.64 Rev. 2 is presented in WCAP-8370 Rev. 8A[1] and WCAP-7800 Rev. 5[2].

Category I

Item	RG 1.68 Rev. 2 9/15/76	Initial Test Programs for Water-Cooled Reactor Power Plants

RESPONSE

Although RG 1.68 Rev. 2 is not within Westinghouse responsibility, but rather the Applicant's, the regulatory guide requirements were considered in the formation of Westinghouse procedures. Westinghouse does not take exception to the guide and is basically in compliance with it's requirements.

Category I

Item

RG 1.84 Rev. 12 3/78 — Code Case Acceptability - ASME Section III Design and Fabrication

RESPONSE

1. Westinghouse controls its suppliers to:

    a. Limit the use of code cases to those listed in Regulatory Position C.1 of the Regulatory Guide 1.84 and 1.85 revision in effect at the time the equipment is ordered, except as allowed in item 2 below.

    b. Identify and request permission for use of any code cases not listed in Regulatory Position C.1 of the Regulatory Guide 1.84 and 1.85 revision in effect at the time the equipment is ordered, where use of such code cases is needed by the supplier.

    c. Permit continued use of a code case considered acceptable at the time of equipment order, where such code case was subsequently annulled or amended.

2. Westinghouse seeks NRC permission for the use of code cases needed by suppliers and not yet endorsed in Regulatory Position C.1 of the Regulatory Guide 1.84 and 1.85 revision in effect at the time the equipment is ordered and permits supplier use only if NRC permission is obtained or is otherwise assured (e.g., a later version of the regulatory guide includes endorsement).

Category I

Item

	RG 1.85 Rev. 12 3/78	Code Case Acceptability - ASME Section III Materials

RESPONSE

See response to RG 1.84, above.

Category I

Item

RG 1.92 Rev. 1 8/22/75	Combining Modal Response and Spatial Components in Seismic Response Analysis

RESPONSE

The Westinghouse procedure for combining modal responses is presented in RESAR-41 Section 3.7.3.4. This procedure is considered an acceptable alternative to RG 1.92 Rev. 1.

Category I

Item

RG 1.100 Rev. 1 6/14/77	Seismic Qualification of Electric Equipment for Nuclear Power Plants

RESPONSE

The Westinghouse program for seismic qualification of safety related electrical equipment to Regulatory Guide 1.100 is delineated in the latest revision of WCAP-8587[3], "Methodology for Qualifying Westinghouse PWR-SD Supplied NSSS Safety Related Electrical Equipment," together with Supplement 1 to this report. In summary, seismic qualification will be demonstrated by the following methods:

1. For equipment not subject to high energy line break conditions which has been previously qualified, as identified in Supplement 1 to WCAP-8587, using the methods permitted by the 1971 version of IEEE Standard 344 (i.e., single axis sine-beat testing or analysis, after demonstration of no resonant frequency below 33 Hz), no additional seismic qualification will be specified, provided that:

   a. It can be shown by separate component testing and/or analysis that there are no aging mechanisms that could prejudice the previously completed seismic qualification.

   b. Any design modifications made to the equipment do not significantly affect the seismic characteristics of the equipment.

   c. The adequacy of the original seismic test levels can be demonstrated as conservative by plant specific verification.

2. For new equipment, or equipment that cannot meet the provisions of 1), above, seismic qualification will be performed in accordance with IEEE Standard 344-75. The method to be employed (i.e., test and/or analysis) is indicated, for the safety related equipment in the Westinghouse PWRSD Scope of Supply, in Supplement 1 to WCAP-8587. Where multifrequency biaxial inputs are employed for testing, the methodology described in WCAP-8695[4], "General Method of Developing Multifrequency Biaxial Test Inputs for Bistables," will be employed. When flexible equipment size and weight precludes biaxial testing (e.g., hydrogen recombiner, enclosures), single axis testing with justification will be utilized to meet IEEE Standard-344 1975. For rigid equipment (i.e., no resonant frequency below 33 Hz), qualification will be by analysis in accordance with IEEE Standard-344 1975.

Category I

Item

RG 1.118 Rev. 1 — 9/27/77 — Periodic Testing of Electric Power and Protection Systems

RESPONSE

ISSUE:

Regulatory Guide 1.118 endorses the use of IEEE-338-1975 as an acceptable method for periodic testing of electric power and protection systems, but supplements the IEEE requirements ("shall" and "must") with the inclusion of recommendations ("should") and further technical considerations not included in the standard.

POSITION:

Westinghouse will treat all "should" statements in IEEE-338-1975 as recommendations to be followed only at its discretion. Detailed positions on the regulatory positions are presented below:

1. Regulatory Position C1

   Westinghouse will provide a means to facilitate response time testing from the sensor input at the protection rack to and including the input to the actuation device. Examples of actuation devices are the protection system relay or bistable.

2. Regulatory Position C2

   Westinghouse defines "Protective Action Systems" to mean the electric, instrumentation and controls portions of those protection systems and equipment actuated and control by the protection system.

3. Regulatory Position C6

   Equipment performing control functions, but actuated from protection system sensors, is not part of the safety system and will not be tested for time response.

4. Regulatory Position C10

   Testing will not be tied to accident conditions, but only to the range of the parameter that is varied.

5. Regulatory Position C11

   Status, annunciating, display, and monitoring functions, except for those related to the Post Accident Monitoring System (PAMS) are considered by Westinghouse to be control functions. Reasonability checks, i.e., comparison between or among similar such display functions, will be made. Otherwise the clarification noted in item 3 above pertaining to Position C6 is observed.

6. Regulatory Positions 12 and 13

Response time testing for control functions operated from protection system sensors will not be performed. Nuclear Instrumentation System detectors will not be tested for time response (see Table 3.3-2, Page 3/4 3-11 of STS). The "expected environmental and mechanical configuration of the actual installation" will not be duplicated for the testing of process sensors which must be removed to accomplish response time testing unless it can be shown that the duplication is practical and that the duplicated factors significantly influence the sensor time response. The Westinghouse-scope protection system does not preclude the response time testing of process sensors by their removal at normal shutdown. The standard Westinghouse-scope protection system does not include design provision which permit insitu testing of process or Nuclear Instrumentation System sensors.

7. Regulatory Position 14

Temporary jumper wires, temporary test instrumentation, the removal of fuses and other equipment not hard-wired into the protection system will be used where applicable.

Category I

Item

	RG 1.123 Rev. 1 7/77	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants

RESPONSE

The Westinghouse position on compliance with Regulatory Guide 1.123 Rev. 1 is presented in WCAP-8370 Rev. 8A[1] and in WCAP-7800 Rev. 5[2].

Category I

Item

RG 1.126 Rev. 0 1/14/77		An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification

RESPONSE

As noted in the discussion in Regulatory Guide 1.126, vendors may utilize densification models other than that noted in the guide, if these models have received NRC approval. As such, Westinghouse utilizes its own densification model as outlined in WCAP-8218[5] (Proprietary) and approved by the NRC.

Category I

Item

NUREG-0102 SRP 1.8 Interfaces for Standard Designs
9/27/77 Rev. 0

RESPONSE

RESAR 41 addresses interfaces in general in Section 1.7. In Section 1.7, an index of acceptable interface information is provided for the balance-of-plant designs, and the applicable interfaces of NUREG-0102 are identified.

Category I

Item

RG 8.19 Rev. 0 3/14/78	Occupational Radiation Dose Assessment at LWRs - Design Stage Man-Rem Estimates

RESPONSE

RESAR 41 does not address Regulatory Guide 8.19. Information needed to address this regulatory guide will be provided in the safety analysis reports referencing the RESAR 41 design.

Category II

Item

RG 1.97 Rev. 1 1/28/77	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident

RESPONSE

The Westinghouse design is in agreement with the regulatory positions of Regulatory Guide 1.97, Revision 1, with the following exceptions:

1. Regulatory Position C.1.

   a. Westinghouse will provide Post Accident Monitoring instrumentation to monitor key Reactor Coolant System parameters, containment conditions, and the effectiveness of the Engineered Safety Features System. The instrumentation provided will provide the operator with information to enable him to perform required manual safety functions and to determine the effect of normal safety actions taken following a reactor trip due to a Condition II, III or IV event.

   b. Instrumentation identified in Battelle-Columbus Laboratories Report BMI-X-647, April 9, 1973, with the exception of that listed in 1.a. above, will not be provided.

2. Regulatory Position C.3

   a. Westinghouse does not supply instrumentation to satisfy position C.3. The delineated parameters in position C.3 extend far beyond the worst case values following SAR, Chapter 15 Design Bases Events.

   b. Post Accident Monitoring instrumentation will be supplied as delineated in item 1.a above. Westinghouse believes that position C.3 should be deleted or modified to maximum range corresponding to worst case conditions. For example, the range for containment pressure is typically 115% of the plant's containment design pressure which is extended to a range that bounds the SAR containment integrity analysis.

3. Regulatory Position C.4

   Post Accident Monitoring instrumentation will be qualified by implementation of the final NRC staff approved version of WCAP-8587[3].

4. Regulatory Position C.5

   a. Westinghouse will provide recorders for certain accident monitoring channels.

b. Of those parameters selected to provide transient or trend information to the operator, at least one of the redundant accident monitoring channels is recorded. The recorder is not redundant, does not meet the single-failure criterion, does not have its own isolation amplifier (the incoming signal will already be isolated from the accident monitoring channel) and may have multiple pens to permit more than one channel to be recorded. The equipment in the generic environmental and seismic qualification program includes these recorders. These recorders will not be qualified to function during the postulated seismic event. Following the event, the recorders will regain an operating status.

5. Implementation

The provisions of this guide will be implemented by Westinghouse with the exceptions noted above.

Category II

Item

RG 1.105 Rev. 1 9/15/76 — Instrument Setpoints

RESPONSE

Westinghouse technical specifications provide the margin from the nominal setpoint to the technical specification limit to account for drift when measured at the rack during periodic testing. The allowances between the technical specification limit and the safety limit include the following items: a) the inaccuracy of the instrument, b) process measurement accuracy, c) uncertainties in the calibration, d) the potential transient overshoot determined in the accident analyses (this may include compensa- tion for the dynamic effect) and e) environmental effects on equipment accuracy caused by postulated or limiting postulated events (only for those systems required to mitigate consequences of an accident). Westinghouse designers choose setpoints such that the accuracy of the instrument is adequate to meet the assumptions of the safety analysis.

The range of instruments is chosen based on the span necessary for the instrument's function. Narrow range instruments will be used where necessary. Instruments will be selected based on expected environmental and accident conditions. The need for qualification testing will be evaluated and justified on a case basis.

Administrative procedures coupled with the present cabinet alarms and/or locks provide sufficient control over the setpoint adjustment mechanism such that no integral setpoint securing device is required. Integral setpoint locking devices will not be supplied.

The assumptions used in selecting the setpoint values in Regulatory Position C.1 and the minimum margin with respect to the technical specification limit and calibration uncertainty will be documented by Westinghouse. Drift rates and their relationship to testing intervals will not be documented by Westinghouse.

Category II

Item

RG 1.124 Rev. 1 8/31/77	Service Limits and Loading Combinations for Class 1 Linear Type Component Supports

RESPONSE

1. The Regulatory Guide states in paragraph B.1(b): "Allowable service limits for bolted connections are derived from tensile and shear stress limits and their study of three interaction curves of allowable tension and shear stress based on the ASME Code (emergency condition allowables per XVII-2110 and faulted condition allowables per F-1370) and the ultimate tensile and shear strength of bolts (obtained from experimental work published by E. Chesson, Jr., N. L. Faustino, and W. H. Munse, "High Strength Bolts Subjected to Tension and Shear," Journal of the Structural Division, Proceedings of the American Society of Civil Engineers, October 1965, Pages 155-180) indicates that there is adequate safety margin between the emergency and faulted condition allowables and failure of the bolts.

   During their tests to determine the strength and behavior characteristics of single high strength bolts subjected to various combinations of tension and shear (T-S), Chesson, et. al. used a total of 115 bolts to ASTM specification A315-61T and A354-Grade BC. The A325-61T, which is a medium carbon steel, had a yield point of 77000 psi to 88000 psi and ultimate strength of 105000 psi to 120000 psi, depending upon the bolt diameter. The A354-Grade BC, which is a heat treated carbon steel, had a yield point of 99000 psi to 109000 psi and ultimate strength from 115000 psi to 125000 psi, depending upon the bolt diameter.

   Figure 3B-1 shows the interaction curves for T-S loads on SA325 bolts. Curve (1) represents the interaction relation (ellipse) permitted by Code Case 1644 (ASME III Appendix XVII Winter 77 Addenda) for service levels A, B and design condition. Curve (2) represents the interaction curve which considers the Code Case 1644 allowables and the increase permitted by XVII-2110(a) for service level C. Curve (3) represents the interaction curve which considers the Code Case 1644 allowables and the increase permitted by F-1370(a) for service level D. Curve (3) is the upper limit of the allowable stresses.

   The design stress limits represented by Curves 1, 2, and 3 for A325 bolts are then compared against the ultimate strength of the bolts represented by Curve 4, which is based on Chesson's test results. The area between Curve 3 and Curve 4 is the safety margin between the maximum bolt stress under service level D and minimum ultimate strength of the bolt.

Factor of safety against failure for A325 bolts for various T-S ratios is shown in Figure 3B-2. The safety factor varies between a minimum of 1.36 and a maximum of 2.29 depending upon the value of T-S ratio. This is based upon the ultimate strength of the bolts from Chesson's test and the allowables obtained from Code Case 1644 and the increase permitted by F-1370(a) for service level D. Figure 3B-2 demonstrates that there exists an adequate factor of safety for the complete range of T-S loadings.

From this study it is observed that:

(1) For the emergency condition, the safety factor (ratio of ultimate strength to allowable stress) varies between a minimum of 1.63 and a maximum of 2.73 depending upon the actual tensile stress/shear stress (T/S) ratio on the bolt.

(2) For the faulted condition, the safety factor varies between a minimum of 1.36 to a maximum of 2.29, again depending upon actual T/S ratio on the bolt.

It is thus reasonable to allow an increase in these limits for the emergency and faulted conditions.

Based on the above discussion for the emergency and faulted conditions, Westinghouse will use allowable bolt stresses specified in Code Case 1644-6, as increased according to the provisions of XVII-2110(a) and F-1370(a), respectively.

2. The increased design limit for the stress range identified in NF-3231.1(a) shall be limited to the smaller of 2 S_Y or S_u unless otherwise justified by shakedown analysis.

3. In paragraphs B.5 and C.8 of the Regulatory Guide, Westinghouse takes exception to the requirement that systems whose safety-related function occurs during emergency or faulted plant conditions must meet level B limits. The reduction of allowable stresses to no greater than level B limits (which in reality are design limits since design level A and level B limits are the same for linear supports) for support structures in those systems with safety related functions occurring during emergency or faulted plant conditions is overly conservative. The primary concern is that the system remains capable of performing its safety function. For active components, this is accomplished through the operability program as discussed in Section 3.9.2.4. In the case of Class 1 piping, maintaining the pipe stresses within level D limits assures that piping geometry is maintained and that required flow is not impeded. The selection of more restrictive stress limits for component supports is not necessary to assure the functional capability of the system.

4. Paragraph C.4 of the Regulatory Guide states: "However, all increases (i.e., those allowed by NF-3231.1(a), XVII-2110(a), and F-1370(a)) should always be limited by XVII-2110(b) of Section III". Paragraph XVII-2110(b) specifies that member compressive axial loads shall be limited to 2/3 of critical buckling.

   In the design of component supports, member compressive axial loads shall be limited to 0.67 times the critical buckling strength. If, as a result of more detailed evaluation of the supports the member compressive axial loads can be shown to safety exceed 0.67 times the critical buckling strength for the faulted condition, verification of the support functional adequacy will be documented and submitted to the NRC for review. The member compressive axial loads will not exceed 0.67 times the critical buckling strength without NRC acceptance. In no case shall the compressive load exceed 0.9 times the critical buckling strength.

5. Paragraph C.4 of the Regulatory Guide states that increases in Level A or B service limits does not apply to limits for bolted connections. The Westinghouse design of component supports restricts the use of bolting material to the following applications:

   a. Westinghouse design uses bolting predominantly in tension. Oversized holes are generally provided and a mechanism other than the bolts is provided to take any shear loads. Shear or shear & tension interaction occur only in isolated locations.

   b. Westinghouse bolts are limited to the following material A490, SA-354, SA-325, SA-540.

   c. The diameters used range between 1/2" and 3".

   These limitations on bolt usage are standard in the Westinghouse supports. We will limit tensile loads in the bolts to 0.7 Su, but not to exceed in any case 0.9 Sy. The allowables are taken at temperature. In those few cases where bolts are used in shear or tension and shear, ASME Code Appendix XVII - 2460 Requirements will apply with an increase factor that is defined in Regulatory Guide 1.124 or in Appendix F-1370, whichever is more restrictive. This provides an adequate margin of safety for the Westinghouse design. If future revisions to the bolting criteria in ASME Section III modify the Westinghouse criteria listed above, we will review the criteria at the time.

6. Paragraph C.6(a) of the Regulatory Guide appears confusing as to what stress limits may be increased for the emergency condition. Westinghouse will interpret this paragraph as follows: "The stress limits of XVII-2000 of Section III and Regulatory Position 3 increased according to the provisions of XVII-2110(a) of Section III and Regulatory Position 4 should not be exceeded for component supports designed by the linear elastic analysis method."

7. The method described in Paragraph C.7(b) of the Regulatory Guide is overly conservative and inconsistent with the stress limits presented in Appendix F. Westinghouse will use the provisions of F-1370(d) to determine service level D allowable loads for supports designed by the load rating method. If future revisions to Appendix F modify this criteria, it will be reviewed further. If the load rating method is used, further details of its implementation will be provided at that time.

Category II

Item

RG 1.130 Rev. 0 7/77	Design Limits and Loading Combinations for Class 1 Plate- and Shell-Type Component Supports

RESPONSE

1. Westinghouse will use the latest revision of Code Case 1644 as approved by Regulatory Guide 1.85.

2. Paragraph B.1 states that increases are not allowed for bolted connections for emergency and faulted conditions. The Westinghouse position is that it is reasonable to allow an increase in the limits for bolted connections for these conditions. Further justification concerning this position can be found in Item 1 of the discussion on Regulatory Guide 1.124.

3. Paragraphs C.3, C.4(a), and C.b(a) state that the allowable buckling strength should be calculated using a design margin of 2 for flat plates and 3 for shells for normal, upset, and emergency conditions.

   In the design of plate-type supports, member compressive axial loads shall be limited per the requirements of Paragraph C.3 for normal upset, and emergency conditions. There are no Class 2 shell-type supports in the Westinghouse NSSS.

4. In paragraph C.7, the inclusion of the upset plant condition in his load combination is inappropriate. The upset plant conditions are properly considered in paragraph C.4.

5. Paragraph C.7(a) references the criterion presented in F-1370(c), which states: "... loads should not exceed 0.67 times the critical buckling strength of the support...".

   In the design of plate-type component supports, member compressive axial loads shall be limited to 0.67 times the critical buckling strength. If, as a result of more detailed evaluation of the supports the member compressive axial loads can be shown to safely exceed 0.67 times the critical buckling strength for the faulted condition, verification of the support functional adequacy will be documented and submitted to the NRC for review. The member compressive axial loads will not exceed 0.67 times the critical buckling strength without NRC acceptance. The Westinghouse NSSS has no Class 1 shell-type supports.

6. The method described in paragraph C.7(b) of the Regulatory Guide is overly conservative and inconsistent with the stress limits presented in Appendix F. Westinghouse will use the provisions of F-1370(d) to determine service level D allowable loads for support designed by the load rating method.

Category II

Item

RG 8.8 Rev. 2 8/18/76	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as Is Reasonably Achievable (Nuclear Power Reactors)

RESPONSE

RESAR 41 does not address Regulatory Guide 8.8 Rev. 2. However, the required infromation to address this regulatory guide is contained in a topical Westinghouse report, WCAP 8872[6], "Design, Inspection, Operation and Maintenance Aspects of the Westinghouse NSSS to Maintain Occupational Exposures As Low As Reasonably Achievable", April 1977. The information in this report will be included in safety analysis reports referencing the RESAR 41 design.

Category III

Item	RG 1.99 Rev. 1 9/15/76	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials

RESPONSE

Justification of the Westinghouse position on Rev. 0 and Rev. 1 of the Guide is detailed in references [7] and [8] respectively.

In summary, Rev. 1 of the Guide is substantially identical with Rev. 0, with minor clarifications and inclusion of a new position C.2, which had previously been included in the Discussion section of Rev. 0.

The Westinghouse letter of comment on Rev. 1 reiterates the comments of Rev. 0 and includes further clarification of hardship imposed by the Guide with respect to vessel material.

The Westinghouse position with respect to each of the Guide positions is as follows:

1. The basis as well as the scope of the Guide for predicting adjustment of reference temperature as given in Regulatory Position C.1 are inappropriate since the data base used was incomplete and included some data which were not applicable.

2. Westinghouse is in agreement with the Guide Position C.2a. However, with respect to Guide Position C.2b, Westinghouse believes that Figure 2 of the Guide is incorrect since the upper shelf energy for six-inch thick ASTM A302B reference correlation monitor material reported by Hawthorne indicates essentially a constant upper shelf at fluences above $\sim 1 \times 10^{19}$ n/cm^2. [8]

3. The Westinghouse position with reference to the Guide Position c.3, controlling residual elements to levels that result in a predicted adjusted reference temperature of less than 200°F at end-of-life, is that the stresses in the vessel can be limited during operation in order to comply with the requirements of Appendix G to 10CFR Part 50 even though the end-of-life adjusted reference temperature may exceed 200°F. By applying the procedures of Appendix G to ASME Section III, the stress limits including appropriate Code safety margin can be met.

4. Recent surveillance capsule data from the Point Beach and Connecticut Yankee reactor vessels [9] indicate a steady state condition of irradiation damage due to the annealing effect at operating temperature. As an alternative to R.G. 1.99, operating limits for Westinghouse plants will be determined by using the current radiation damage curves developed by Westinghouse [10]. It is expected that, as more surveillance capsule data is accumulated, both the R.G. 1.99 and Westinghouse damage curves will prove to be overly conservative.

Category III

Item

RG 1.121 Rev. 0 5/11/76 — Bases for Plugging Degraded PWR Steam Generator Tubes

RESPONSE

Position C.1

Westinghouse interprets the term "Unacceptable defects" to apply to those imperfections resulting from service induced mechanical or chemical degradation of the tube walls which have penetrated to a depth in excess of the Plugging Limit.

Positions C.2.a.(2) and C.2.a(4)

Westinghouse has documented its opinions on Reg. Guide 1.121 by corporate letter and has identified as the major exception the margin of 3 against tube failure for normal operation. Westinghouse defines tube failure as plastic deformation of a crack to the extent that the sides of the crack open to a non-parallel, elliptical configuration. The tubing can sustain added internal pressure beyond those values before reaching a condition of gross failure. We have interpreted this to apply as an operating limit for the plant and consider that it introduces a conflict to the established conditions for plant operation as identified in the plant tech specs. A factor of 3 is quite often used in ASME Code Design guidelines. These Code practices apply to the design of hardware and to the analyses done on these designs. Conditions which occur during operation of the equipment and which may affect the equipment so that design values no longer apply, are not directly addressed by the initial Code requirements. That is one reason why plant tech specs have been generated to establish safe limits of operation for power station equipment. The ASME Code is not applicable to the operational criteria of steam generator tubing. Our tubing design and tubing in the design condition has margins in excess of 3. In summary, we satisfy the margin of 3 if it were used in a Code sense as new equipment design. Moreover, we do not believe that this margin should be utilized as a limiting conditions for normal operation.

Position C.2.b

In cases where sufficient inspection data exists to establish a degradation allowance, the rate used will be an average time-rate determined from the mean of the test data.

Position C.3.d(1) and C.3.d.(3)

The combined effect of these requirements would be to establish a maximum permissable primary-to-secondary leak rate which may be below the threshold of detection with current methods of measurement. Westinghouse has determined the maximum acceptable length of a through-wall-crack based on secondary pipe break accident loadings which are typically twice the magnitude of normal operation pressure loads. Westinghouse will use a leak rate associated with the crack size determined on the basis of accident loadings.

Position C.3.e.(6)

Westinghouse will supply computer code names and references rather than the actual codes.

Position C.3.f.(1)

Westinghouse will establish a minimum acceptable tube wall thickness (Plugging Limit) based on structural requirements and consideration of loadings, measurement accuracy and, where applicable, a degradation allowance as discussed in this position and in accordance with the general intent of this guide. Analyses to determine the maximum acceptable number of tube failures during a postulated condition are normally done to entirely different bases and criteria are not within the scope of this guide.

Position C.3.f.(4)

Where requirements for minimum wall are markedly different for different areas of the tube bundle, e.g., U-bend area versus straight length in Westinghouse designs, two plugging limits may be established to address the varying requirements in a manner which will not require unnecessary plugging of tubes.

Category III

Item

SRP 5.4.7 Rev. 1 1/31/78 — Residual Heat Removal System

RESPONSE

Based on SRP 5.4.7 and Branch Technical Position RSB 5-1, the following technical requirements are applicable to RESAR 41 plants:

1. Provide safety-grade steam generator dump valves, operators, air and power supplies, which meet the single failure criterion.

2. Provide the capability to cooldown to cold shutdown in less than 36 hours assuming the most limiting single failure and loss of offsite power or show that manual actions inside or outside containment or return to hot standby until the manual actions or maintenance can be performed to correct the failure provides an acceptable alternative.

3. Provide the capability to depressurize the reactor coolant system with only safety-grade systems assuming a single failure and loss of offsite power or show that manual actions inside or outside containment or remaining at hot standby until manual actions or repairs are complete provides an acceptable alternative.

4. Provide the capability for borating with only safety-grade systems assuming a single failure and loss of offsite power or show that manual actions inside or outside containment or remaining at hot standby until manual action or repairs are completed provides an acceptable alternative.

5. Provide the system and component design features necessary for the prototype testing of both the mixing of the added borated water and the cooldown under natural circulation conditions with and without a single failure of a steam generator atmospheric dump valve. These tests and analyses will be used to obtain information on cooldown times and the corresponding AFW requirements.

6. Commit to providing specific procedures for cooling down using natural circulation and submit a summary of these procedures.

7. Provide or require a seismic Category I AFW supply for at least 4 hours at Hot Shutdown plus cooldown to the DHR system cut-in based on the longest time (for only onsite or offsite power and assuming the worst single failure), or show that an adequate alternate seismic Category I source will be available.

The safe shutdown design basis for all Westinghouse pressurized water reactors is hot standby. Following any Condition II, III, or IV event including loss of offsite power, RESAR 41 plants can be placed in and maintained in, for an extended period of time, a safe hot standby condition using safety grade systems only. In this condition, decay heat removal is via the auxiliary feedwater system and the steam generator safety valves. Westinghouse recommends that the plant be maintained in the safe hot standby condition following a Condition II, III, or IV event. However, Westinghouse pressurized water reactors are capable of being taken to cold shutdown, if required, provided that restrictions are not placed on operator manual actions.

The technical requirements of BTP RSB 5-1 are addressed as follows:

1. Compliance is not required since the plant can be maintained in a safe hot standby condition while any required manual actions are taken. One safety grade steam generator power operated relief valve is provided for each of the four steam generators. Safety grade remote operators and power supplies are not required since hot standby can be achieved and maintained using the safety grade steam generator safety valves. The steam generator power operated relief valves are provided with handwheels and can be operated locally to permit plant cooldown.

2. Compliance is not required since the plant can be maintained in a safe hot standby condition while any required manual actions are taken. The plant is capable of reaching RHR initiation conditions in approximately 36 to 48 hours, including the time required to perform any manual actions.

3. Compliance is not required since the plant can be maintained in a safe hot standby conditions while any required manual actions are taken.

4. Compliance is not required since the plant can be maintained in a safe hot standby condition while any required manual actions are taken.

5. The plant design provides the capability for conducting natural circulation cooldown tests if required. However, other Westinghouse designed pressurized water reactors will have conducted such tests prior to operation of future RESAR 41 plants. Because of the great similarity in design between all Westinghouse pressured water reactors, previously conducted tests can be referenced in lieu of conducting such tests on every unit.

6. Specific procedures for cooling down using natural circulation will be prepared prior to start up.

7. Sufficient auxiliary feedwater is provided in a seismic Category I supply to permit four hours operation at hot standby plus cooldown to RHR initiation conditions, including allowance for time to correct single failures which might occur.

Category III

Item

RG 1.141 Rev. 0 1/31/78 — Containment Isolation Provisions for Fluid Systems

RESPONSE

This Regulatory Guide describes a method acceptable to the NRC Staff for complying with the commission requirements with respect to containment isolation. The guide basically endorses the requirements of ANSI N271-1976 for the design of containment isolation provisions for fluid systems. However, several exemptions and changes to the provision of the standard are also presented.

Westinghouse's containment isolation philosophy for fluid systems complies with the guidance provided by ANSI N271-1976 and/or Regulatory Guide 1.141 with the following exceptions and/or clarifications.

1. The Standard in Section 3.6.3 states that remote manual closure of isolation valves on ESF or ESF related systems is acceptable when provisions are made to detect possible failure of the fluid lines inside and outside containment. Although such provisions are outside Westinghouse scope of supply, Westinghouse is of the opinion that provisions to detect failure of fluid lines inside containment are unnecessary. Since redundant ESF capacity is provided and offsite doses due to leakage inside containment are not a concern, Westinghouse does not require or provide for detection of failures in fluid lines inside containment.

   Section 3.6.4 states that a single valve and closed system outside containment isacceptable if the closed system is treated as an extension of the containment. Further, the standard requires that the valve and the piping between the valve and the containment be enclosed in a protective leak tight or controlled leakage compartment. The closed system is also required to be leak tested in accordance with 10 CFR 50 Appendix J unless it can be shown by inspection that system integrity is being maintained for those systems operating during normal plant operation at a pressure equal to or above the containment design pressure.

   Westinghouse employs this design arrangement on the ECCS sump isolation valves and is in basic agreement with the provisions of the standard. However, in the case of the sump isolation valves, Westinghouse perceives no basis for the requirement to leak test the closed system. The recirculation system (closed system), regardless of the sump isolation configuration will be circulating radioactive fluid during LOCA conditions. Should a leak develop in a recirculation loop, that loop can be isolated by remote closure of the sump isolation valve to prevent further loss of sump water. Should a leak develop in the isolation valve body or in the piping between the sump and the valve, then the sump fluid will be contained by the leaktight, compartment and guardpipe arrangement. (Westinghouse design does make provision for compartment/guard pipe leak testing) With these provisions no single active or passive failure will prevent the recirculation of core cooling water or adversely affect the integrity of the containment.

3. Section 4.2.5 and 4.4.6 of the standard as implemented by item C3 of the regulatory guide are intepreted by Westinghouse to state that: to preclude common mode failures, diversity is required in the parameters sensed from which isolation signals are generated. Westinghouse design criteria for the initiation of containment isolation does not include a requirement for diversity in the primary station variable for any given design condition or event. Westinghouse, however, utilizes different primary station variables to derive and generate the protective function for the first phase (A) of containment isolation. Some diversity is therefore available for this phase for a given event. The second phase of containment isolation(B), which isolates only component cooling water to the reactor coolant pump, is initiated only by a high containment pressure signal. Diversity is therefore not available for this containment isolation function, nor is it believed necessary by Westinghouse.

4. The standard states in Section 1 that "If an accident occurred, fluid systems penetrating the containment would be isolated except those which are engineered safety related." With respect to this recommendation the following clarification is provided. The isolation valves in the reactor coolant pump seal injection lines will be tripped closed on a T signal coincident with low charging pump header pressure. Thus, positive isolation is ensured via isolation valve closure or fluid inflow.

Category III

Item

RSB 5-2 Rev. 0 3/14/78 — Reactor Coolant System Overpressurization Protection

RESPONSE

The pressurizer power-operated relief valves will be supplied with additional actuation logic to ensure that a completely automatic and independent RCS pressure control backup feature is provided for the operator during low-temperature operations. This system provides the capability for additional RCS inventory letdown, to minimize the number of occurrences of pressure transients and to reduce the severity of such transients, should they occur.

The basic function of the system logic is to continuously monitor RCS temperature conditions, with the logic armed whenever plant operation is at low temperatures. An auctioneered system temperature will be continuously converted to an allowable pressure and then compared to the actual RCS pressure. This comparison, when required, will provide an actuation signal to the power-operated relief valves.

Category IV-A

Item

RG 1.14 Rev. 1 8/75 — Reactor Coolant Pump Flywheel Integrity

RESPONSE

Since the issuance of Regulatory Guide 1.14, Revision 1, the NRC Staff has provided to Westinghouse a copy of Draft 2, Revision 2 of Regulatory Guide 1.14 (via on April 12, 1976 letter from Robert B. Minogne to C. Eicheldinger). This draft was formulated from industry and concerned parties' comments. It is significant that the Draft 2 version incorporates several of the Westinghouse comments on Revision 1. Since Draft 2 has not been formally published as Revision 2 of Regulatory Guide 1.14, the exceptions and clarifications (from the original Westinghouse comments) are provided below:

1. Post-Spin Inspection

   Westinghouse has shown in WCAP-8163, "Topical Report Reactor Coolant Pump Integrity in LOCA,"[11]"that the flywheel would not fail at 290% of normal speed for a flywheel flaw of 1.15 inches or less in length. Results for a double ended guillotine break at the pump discharge with full separation of pipe ends assumed, show the maximum overspeed to be less than 110% of normal speed. The maximum overspeed was calculated in WCAP-8163 to be about 280% of normal speed for the same postulated break, and an assumed instantaneous loss of power to the reactor coolant pump. In comparison with the overspeed presented above, the flywheel could withstand a speed up to 2.3 times greater than the flywheel spin test speed of 125% provided that no flaws greater than 1.15 inches are present. If the maximum speed were 125% of normal speed or less, the critical flaw size for failure would exceed 6 inches in length. Non-destructive tests and critical dimension examinations are all performed before the spin tests. The inspection methods employed (described in WCAP-8163) provide assurance that flaws significantly smaller than the critical flaw size of 1.15 inches for 290% of normal speed would be detected. Flaws in the flywheel will be recorded in the pre-spin inspection program (see WCAP-8163). Flaw growth attributable to the SPIN test (i.e., from a single reversal of stress, up to speed and back), under the most adverse conditions, is about three orders of magnitude smaller than what non-destructive inspection techniques are capable of detecting. For these reasons, Westinghouse performs no post-spin inspection and believes that pre-spin test inspections are adequate.

2. Interference Fit Stresses and Excessive Deformation

   Much of Revision 1 deals with stresses in the flywheel resulting from the interference fit between the flywheel and the shaft. Because Westinghouse's design specifies a light interference fit between the flywheel and the shaft; at zero speed, the hoop stresses and radial stresses at the flywheel bore are negligible. Centering of the flywheel relative to the shaft is accomplished by means of keys and/or centering devices attached to the shaft, and at normal speed, the flywheel is not in contact with the shaft in the sense intended by Revision 1. Hence, the definition of "Excessive Deformation," as defined in Revision 1 of Regulatory Guide 1.14, is not applicable to the Westinghouse designs since the enlargement of the bore and subsequent partial separation of the flywheel from the shaft does not cause unbalance of the flywheel. Extensive Westinghouse experience with reactor coolant pump flywheels installed in this fashion has verified the adequacy of the design.

Westinghouse's position is that combined primary stress levels, as defined in Revision 0 of Safety Guide 14 (C.2) (a) & (c) are both conservative and proven and that no changes to these stress levels are necessary. Westinghouse designs to these stress limits and thus does not have permanent distortion of the flywheel bore at normal or spin test conditions.

3. Section B, Discussion of Cross Rolling Ratio of 1 to 3

Cross Rolling Ratio - Westinghouse's position is that specification of a cross rolling ratio is unnecessary since past evaluations have shown that ASME SA-553-B Class 1 materials produced without this requirement have suitable toughness for typical flywheel applications. Proper material selection and specification of minimum material properties in the transverse direction adequately ensure flywheel integrity. An attempt to gain isotropy in the flywheel material by means of cross rolling is unnecessary since adequate margins of safety are provided by both flywheel material selection (ASME SA-533-B Class 1) and by specifying minimum yield and tensile levels and toughness test values taken in the direction perpendicular to the maximum working direction of the material.

4. Section C, Item 1a relative to vacuum-melting and degassing process or the electroslag process

Vacuum Treatment - The requirements for vacuum melting and degassing process or the electroslag process are not essential in meeting the balance of the Regulatory Position nor do they, in themselves, ensure compliance with the overall Regulatory Position. The initial Safety Guide 14 stated that the "flywheel material should be produced by a process that minimized flaws in the material and improves its fracture toughness properties." This is accomplished by using SA-533 material including vacuum treatment.

5. Section C, Item 2b/ Westinghouse interprets this paragraph as delineated below. This removes the ambiguous reference to an undefined overspeed transient.

Design Speed Definition

Design speed should be 125% of normal speed or the speed to which the pump motor might be electrically driven by the station turbine generator during anticipated transients, whichever is greater. Normal speed is defined as the synchronous speed of the a-c drive motor at 60 Hz.

Category IV-A

Item

RG 1.75 Rev. 0 1/75 — Physical Independence of Electric Systems

RESPONSE

The physical separation criteria for redundant safety-related system sensors, sensing lines, wireways, cables, and components on racks within Westinghouse NSSS scope meet recommendations contained in Regulatory Guide 1.75 with the following comments:

a. The Westinghouse design of the protectior system relies on the provisions of IEEE Standard 384-1974 re'ative to overcurrent devices to prevent malfunctions to one circuit from causing unacceptable influences on the functioning of the protection system. The protection system uses redundant instrumentation channels and actuation trains and incorporates physical and electrical separation to prevent faults in one channel from degrading any other protection channel.

b. Separation recommendations for redundant instrumentation racks are not the same as those given in Regulatory Position C.1 of Regulatory Guide 1.75, Revision 1, for the control boards because of different functional requrements. Main control boards contain redundant circuits which are required to be physically separated from each other. However, since there are no redundant circuits which share a single compartment of an NSSS protection ir trumentation rack, and since these redundant protection instru tation racks are physically separated from each other, the ical separation requirements specified for the main control board do not apply.

However, redundant, isolated control signal cables leaving the protection racks are brought into close proximity elsewhere in the plant, such as the control board. It could be postulated that electrical faults, or interference, at these locations might be propagated into all redundant racks and degrade protection circuits because of the close proximity of protection and control wiring within each rack. Regulatory Guide 1.75 (Regulatory Position C.4) and IEEE Standard 384-1974 (Section 4.5(3)) provide the option to demonstrate by tests that the absence of physical separation could not significantly reduce the availability of Class 1E circuits.

Westinghouse test programs have demonstrated that Class 1E protection systems (nuclear instrumentation system, solid state protection system and 7300 process control system) are not degraded by non-Class 1E circuits sharing the same enclosure. Conformance to the requirements of IEEE Standard 279-1971 and Regulatory Guide 1.75 has been established and accepted by the NRC based on the following which is applicable to these systems at the Seabrook sites.

Tests conducted on the as-built designs of the nuclear instrumentation system and solid state protection system were reported and accepted by the NRC in support of the Diablo Canyon application (Docket Nos. 50-275 and 50-323). Westinghouse considers these programs as applicable to all plants. Westinghouse tests on the 7300 process control system were covered in a report entitled, "Westinghouse 7300 Series Process Control System Noise Tests," subsequently reissued as Reference 14. In a letter dated April 20, 1977 (Reference 15) the NRC accepted the report in which the applicability to RESAR-41 is established.

c. The physical separation criteria for instrument cabinets within Westinghouse NSSS scope meet the recommendations contained in Section 5.7 of IEEE Standard 384-1974.

Category IV-A

Item

RG 1.79 Rev. 1 9/75	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors

RESPONSE

This regulatory guide is not within Westinghouse scope. However, the Westinghouse test program, as outlined in Chapter 14, conforms to the requirements of this guide with the exception of the three items noted below. In addition Westinghouse does assist the applicant whenever the latter desires assistance.

Westinghouse takes exception to Regulatory Guide 1.79, Revision 1, as follows:

1. High Pressure Safety Injection (HPSI) Flow Test

   The guide specifies that the cold and hot safety injection flow test should be initiated by the safety injection signal with all affected auxiliary systems in their standard operating mode.

   The Westinghouse safety injection flow test procedures do not require the use of a safety injection signal to initiate safety injection flow. Other startup test procedures check the operation of circuit breakers, controls, logic and interlocks for the safeguard systems.

   Considering the above and that the use of a safety injection signal would require the racking out of various circuit breakers for the injection flow tests to negate unwanted operations, Westinghouse does not specify the use of a safety injection signal for the injection flow tests.

2. Low Pressure Safety Injection (LPSI) Recirculation Test

   The Regulatory Guide specifies that recirculation should be demonstrated by taking suction from the containment floor or sump. The Westinghouse test program recommends that suction be taken from the RWST. The net positive suction head can be determined from the RWST level and the level in the containment sump and shown to be greater than the required net positive suction head for the pump. The design of the containment sump and screens are outside Westinghouse scope, and therefore, the verification of vortex and control and pressure drop across the screens would be peculiar to a particular plant.

3. Core Flooding Flow Test - Hot Operating Conditions

The intent of the test is to verify that check valves that see higher than ambient temperatures during power operation will function at the higher temperatures.

The regulatory guide specifies slowly decreasing RCS pressure and temperature until the accumulator check valves operate as indicated by a decrease in accumulator level.

The Westinghouse test program utilizes the high head safety injection pumps and the check valve leakage test lines to provide flow through the check valves. Operation of the check valves would be noted by an increase in pressurizer level. The Westinghouse method allows the check valve to be nearer to the operating temperature as a result of the RCS being near operating pressure and temperature versus the lowering the pressure and temperature per the Regulatory Guide. In addition, the Westinghouse test program recommends that the remaining safety injection system check valves that would be subjected to RCS temperatures also be checked as described above.

Category IV-A

Item

RG 1.83 Rev. 1 7/75	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes

RESPONSE

Access

The Westinghouse steam generator design permits access to all of the tubes for inspection, plugging, or other repair.

Baseline Inspection

Westinghouse concurs with the option of the last paragraph of Section B which permits the shop examination of tubing to serve as an adequate baseline inspection, provided that the examination is done in accordance with the requirements of the ASME Code Section III, Subsection NB, Article 2550. The owner may at his option perform the inspection prior to operation of the plant in accordance with paragraph C.3.a.

Sample Selection, Testing and Acceptance Limits

The details of the inservice inspection are not within the scope of supply of the Nuclear Steam Supply System.

Category IV-A

Item

RG 1.89 Rev. 0 11/74	Qualification of Class 1E Equipment for Nuclear Power Plants

RESPONSE

To meet the requirements of Regulatory Guide 1.89 Westinghouse intends to implement a practical interpretation of IEEE Standard 323-1974 which is mutually acceptable to the NRC and Westinghouse. This interpretation is presently being developed; and the NRC staff is being consulted and kept informed with regard to this program.

Category IV-B

Item	Branch	Applicable SRP Section	Title
1 11/24/75	MTEB	5.4.2.1	BTP MTEB-5-3 Monitoring of Secondary Side Water Chemistry in PWR Steam Generator

RESPONSE

As noted in Section 10.1 of RESAR-41, the water chemistry of the steam side and its effectiveness in corrosion control will be discussed in Chapter 10 of the applicants Safety Analyses Report. Recommendations by Westinghouse to the applicant on monitoring of secondary side water chemistry are in compliance with BTP MTEB-5-3.

Category IV-B

Item	Branch	Applicable SRP Section	Title
2 11/24/75	CSB	6.2.1, 6.2.1A 6.2.1B 6.2.1.2 6.2.1.3 6.2.1.4 6.2.1.5	BTP CSB-6-1, Minimum Containment Pressure Model for PWR ECCS Performance Evaluation

RESPONSE

Westinghouse analyses are in compliance with SRP 6 ?.1.5, BTP CSB-6-1, Minimum Containment Pressure Model for PWR ECCS Evaluation.

Category IV-B

Item	Branch	Applicable SRP Section	Title
3 11/24/75	CSB	6.2.5	BTP CSB-6-2, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident.

RESPONSE

Equipment to maintain control of post LOCA hydrogen within the limits specified in Branch Technical Position CSB-6-2 is an optional scope item not covered by RESAR 41. When this option is exercised, Westinghouse provides a hydrogen recombiner unit whose design, operating limits, and seismic qualification is in compliance with BTP CSB-6-2. RESAR-41 Section 15.4.1.2 provides the appropriate interface information per Regulatory Guide 1.7 for the applicant in meeting the requirements of BTP CSB-6-2. (See response in Category I to RG 1.7)

Category IV-B

Item	Branch	Applicable SRP Section	Title
5 11/24/75	CSB	6.2.4	BTP CSB-6-4, Containment Purging During Normal Plant Operations

RESPONSE

Containment purging, in compliance with SRP Section 6.2.4, is the responsibility of the utility and/or AE. Hydrogen production rates used to establish purge requirements are supplied as interface information in RESAR 41. This hydrogen production is calculated per Reg. Guide 1.7.

Category IV-B

Item	Branch	Applicable SRP Section	Title
6 11/24/75	ASB	9.1.4	BTP ASB-9.1, Overhead Handling Systems for Nuclear Power Plants

RESPONSE

Westinghouse is in compliance with SRP Section 9.1.4, BTP-ASB 9.1 "Overhead Handling Systems for Nuclear Power Plants" for RESAR-41. Although the overhead crane is not in Westinghouse scope, Westinghouse has performed the required dropped vessel head analysis for RESAR-41, and this analysis was accepted per NRC letter no. STN 50-480 of November 30, 1976.

Category IV-B

Item	Branch	Applicable SRP Section	Title
8 11/24/75	SEB	3.5.3	Procedures for Composite Section Local Damage Prediction (SRP Section 3.5.3, par. II.1.C)

RESPONSE

Composite missile barriers are not within Westinghouse scope. The only missile barrier provided is a steel CRDM missile shield for the integrated head, as discussed in RESAR-41, Section 3.5.4.

Category IV-B

Item	Branch	Applicable SRP Section	Title
10 11/24/75	SEB	3.7.2	Procedures for Seismic System Analysis (SRP Section 3.7.2 par. II)

RESPONSE

Table 3B-2 identifies the RESAR-41 sections where the acceptance criteria of SRP 3.7.3 are addressed. Note that SRP 3.7.3.8 is not in Westinghouse scope. Interactions of non-Category I structures with Category I structures are appropriate for coverage in the Applicant's Safety Analysis Report.

Category IV-B

Item	Branch	Applicable SRP Section	Title
11 11/24/75	SEB	3.7.3	Procedures for Seismic Subsystem Analysis (SRP Section 3.7.3 par., II)

RESPONSE

A matrix which identifies the RESAR-41 section where the acceptance criteria of SRP 3.7.3 are addressed is given in Table 3B-1. The Westinghouse position on SRP 3.7.3.5, which is not addressed in RESAR-41, follows

3.7.3.5 Use of Equivalent Static Load Method of Analysis

The equivalent static load or static analysis method involves the multiplication of the total weight of the equipment or component member by the specified seismic acceleration coefficient. The magnitude of the seismic acceleration coefficient is established on the basis of the expected dynamic response characteristics of the component. Components which can be adequately characterized as single degree of freedom systems are considered to have a modal participation factor of one. Seismic acceleration coefficients for multi-degree of freedom systems which may be in the resonance region of the amplified response spectra curves are increased by 50 percent to account conservatively for the increased modal participation.

Category IV-B

Item	Branch	Applicable SRP Section	Title
14 11/24/75	SEB	3.8.3	Structural Design Criteria for Category I Structures Inside Containment (SRP Section 3.8.3, par. II).

RESPONSE

The applicant has responsibility for concrete and steel structures inside containment with the exception of supports for the reactor pressure vessel, steam generators, reactor coolant pumps, pressurizer, and loop piping. Design criteria for the supports within Westinghouse scope are addressed in RESAR-41, Section 5.5.1.4.

Category IV-B

Item	Branch	Applicable SRP Section	Title
17 11/24/75	SEB	3.7 11.2 11.3 11.4	Seismic Design Requirements for Radwaste systems and their Housing Structures (SRP Section 11.2, BTP ETSB 11-1, par. B.[illegible])

RESPONSE

Westinghouse Waste Processing Systems meet the requirements of Regulatory Guide 1.143. These guidelines are spelled out in SRP Section 11.2, and BTP ETSB 11.1. However, the structures and enclosures housing the waste systems are not within Westinghouse scope. Westinghouse has and will continue to provide all necessary interface information to A/E and utilities.

Category IV-B

Item	Branch	Applicable SRP Section	Title
20 11/24/75	ASB	10.4.7	Waterhammer for Steam Generators with Preheaters (SRP 10.4.7 par. I.2.b)

RESPONSE

The design of the RESAR-41 steam generators with preheaters eliminates the possibility of waterhammer during normal operations. This was confirmed by a 1/8 scale model test of the steam generator preheat section. The results of this test were submitted in WCAP-9232 [12], "High Pressure Water Hammer Test of the Split Flow Preheat Steam Generator."

Since the conditions necessary for water hammer could occur during certain accident conditions, WCAP 9232 also summarizes the results of a structural analysis of the steam generator tubes. The structural analysis results show that even under the combined loadings of a main steam line break and a water hammer event, steam generator tube integrity is maintained.

Category IV-B

Item	Branch	Applicable SRP Section	Title
21 11/24/75	AB	4.4	Thermal-Hydraulic Stability (SRP Section 4.4, par. II.5)

RESPONSE

At this time Westinghouse is not in full compliance with the requirements of Section 4.4 par. II.5. However, Westinghouse does not take exception to the Section, but rather has discussed the issue with the NRC and is taking the steps necessary to secure NRC approval of the Westinghouse methods.

Category IV-B

Item	Branch	Applicable SRP Section	Title
22 11/24/75	RSB	5.2.5	Intersystem Leakage Detection (SRP Section 5.2.5 par II.4) and RG 1.45

RESPONSE

Reliable methods of intersystem leakage detection for systems connected to the RCS are provided in the RESAR 41 design. Leakage into the RHRS or SIS (the primary NSSS systems of concern) would be directly detectable by design provisions of both systems. For leakage into piping systems which include surge volumes (e.g., accumulator, or RWST) the leakage would be detected by such means as increasing level or pressure indications and/or alarms. For systems which are water solid leakage would be detected by lifting of relief valves with, depending on the location, increased level and temperature indication in the relief volume.

The RHRS and SIS are in all cases isolated from the RCS by at least two normally closed valves in series each of which is designed to have zero seat leakage. Where appropriate these isolation valves comply with requirements of Branch Technical Position ICSB 3 "Isolation of Low Pressure from High Pressure Reactor Coolant System." Further, it should be noted that stringent limitations require RCS leakage to be within limits specified in the Applicants Technical Specifications or require the plant to be placed in cold shutdown.

Category IV-C

Item	Branch	Applicable SRP Section	Title
2 8/1/76	SEB	3.7.1	Response Spectra in Vertical Direction

RESPONSE

The design response spectra of Regulatory Guide 1.60 Rev. 1 are acceptable to Westinghouse with the following exceptions:

a. The response spectrum for vertical motion is taken as 2/3 the response spectrum for horizontal motion over the entire range of frequencies in the western United States.

b. The damping values recommended and approved by the Staff in WCAP-7921-AR[13] "Damping Values of Nuclear Power Plant Components" are used in dynamic analysis of Westinghouse supplied equipment.

Category IV-C

Item	Branch	Applicable SRP Section	Title
6 6/11/77	RSB	6.3	Passive Failures During Long-Term Cooling Following LOCA

RESPONSE

Westinghouse philosophy on passive failure meets the intent of the General Design Criteria on single failure as it specifically applies to failure of passive components in the ECCS. The complete Westinghouse position on passive failure criteria may be found in RESAR-41, Section 6.3.2.11.

The proposed NRC position requires:

1. Identification and justification of maximum leak rate. This information can be provided. A review of the equipment in the ECCS indicates that the largest sudden leak potential would be the sudden failure of a pump shaft seal. Evaluation of leak rates showed that flows less than 50 gpm would result. The other items presented in the NRC position fall into the A/E's scope.

Category IV-C

Item	Branch	Applicable SRP Section	Title
7 9/1/77	RSB	6.3	Control Room Position Indication of Manual (Handwheel) Valves in the ECCS

RESPONSE

Manual Valves

Manual valves in the SIS are mainly those required for maintenance, refueling, or test operations. Those valves that, if improperly positioned, would have an adverse effect on the performance of the ECCS are physically locked in the correct position. Manual throttle valves in the injection branch lines are properly adjusted by flow tests during initial startup testing. The operating handles on these valves are then removed and the stems covered with a locked cover. The manual valves remain locked (either open or closed), therefore, control room position indication is not necessary.

Category IV-C

Item	Branch	Applicable SRP Section	Title
8 4/1/77	RSB	15.1.5	Long-Term Recovery from Steamline Break: Operator Action to Prevent Overpressurization

RESPONSE

For long term cooling following a steamline break, the operator is instructed to use the intact steam generators for the purpose of removing decay heat and plant stored energy. This is done by feeding the steam generators with emergency feedwater to maintain an indicated water level in the steam generator narrow-range span.

Steam pressure from the steam generators is relieved by the main steam safety valves or the steam generator power relief valves. The operator is instructed to terminate emergency feedwater flow to the faulted steam generator as soon as he determines which steam generator is faulted. As soon as an indicated water level returns to the pressurizer, the operator is instructed to turn off the charging pumps to limit system repressurization.

Following the hypothetical steamline break incident, a steamline isolation signal will be generated almost immediately, causing the steamline isolation valves to close within a few seconds. If the break is downstream of the isolation valves, all of which subsequently close, the break will be isolated. If the break is upstream of the isolation valves or one valve fails to close, the break will be isolated from three steam generators while the faulted one will continue to blow down. Only the case in which one steam generator continues to blow down is discussed here since the break followed by isolation of all steam generators will terminate the transient.

A safety injection signal (generated a few seconds after the break) will cause main feedwater isolation to occur. The only source of water available to the faulted steam generator is then the emergency feedwater system. Following steamline isolation, steam pressure in the steamline with the faulted steam generator will continue to fall rapidly, while the pressure stabilizes in the remaining two steam lines. The indication of the different steam pressures will be available to the oper. within a few seconds of steamline isolation. This will provide the information necessary to identify the faulted steam generator so that emergency feedwater to it can be isolated. Manual controls are provided in the control room for start and stop of the emergency feedwater pumps and the control valves associated with the emergency feedwater system. The means of detecting the faulted steam generator and isolating emergency feedwater to it requires only the use of safety grade equipment available following the break. The removal of decay heat in the long term (following the initial cooldown) using the remaining steam generators requires only the emergency feedwater system as a water source and the secondary system safety valves to relieve steam.

The operator has available, in the control room, an indication of pressurizer water level from the instrumentation used in the reactor protection system. Indicated water level returns to the pressurizer in approximately five to seven minutes following the steamline break. To maintain the indicated water level, the operator can start and stop the charging pumps as necessary. The pressurizer level instrumentation and manuals controls for operation of the charging pumps meet the required standards for safety systems.

As indicated, the information for terminating emergency feedwater is available to the operator within one minute of the break while the information required for stopping the charging pumps becomes available within five to seven minutes following the break. The requirements to terminate emergency feedwater flow to the faulted steam generator and stop the charging pumps can be met by simple switch actions by the operators, i.e., closing emergency feedwater discharge valve and stopping the charging pumps. Thus, the required simple actions to limit the cooldown and repressurization can be easily recognized, planned and performed within ten minutes. The time at which operator action is required to prevent the pressure-temperature limits from being exceeded is in excess of ten minutes. For decay heat removal and plant cooldown the operator has a considerably longer time period in which to respond because of the large initial cooldown associated with steamline break transient.

The safety related indicators for steamline pressure and pressurizer water level noted above are further discussed in Section 7.5.

Category IV-C

Item	Branch	Applicable SRP Section	Title
9 12/1/77	RSB	5.4.6 5.4.7 6.3	Pump Operability Requirements

RESPONSE

The Equipment Specification for the safety injection pumps to be employed on RESAR-41 applications will require them to be capable of performing their long term cooling function for one year. These pumps will be used on the South Texas Project and have undergone development testing. Testing included endurance testing and combined crud/thermal transient testing. Satisfactory results from these tests have confirmed the long term acceptability of the SI pump designs.

Category IV-C

Item	Branch	Applicable SRP Section	Title
10 3/28/78	SRP	3.5.1	Gravity Missiles, Vessel Seal Ring Missiles Inside Containment

RESPONSE

Gravity Missiles

The evaluation of the potential for gravity missiles is a function of plant layout and, therefore, not in the Westinghouse NSSS scope.

Reactor Vessel Cavity Seal Ring Missiles

The reactor vessel cavity seal ring is designed to withstand a pressure from underneath as specified in each individual plant design. During LOCA conditions, depending upon the specific plant design and layout, the reactor vessel cavity seal ring may be a potential missile. The evaluation of the effect of this potential missile on plant safety will be discussed in the Applicant's Safety Analysis Report.

Category IV-C

Item	Branch	Applicable SRP Section	Title
11 1/1/77	AB	4.4	Core Thermal-Hydraulic Analysis

NRC Position

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

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

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

3. The effect of fuel rod bowing.

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

RESPONSE

Westinghouse is in compliance with items 1, 2, and 3 (effect of radial pressure gradients at exit of open lattice cores and the upper plenum, and effect of fuel rod bowing) in that these effects have been noted in the Westinghouse evaluation of the thermal-hydraulic performance of the core.

At this time, Westinghouse does not intend to perform special tests to verify the transient analysis methods and codes. Discussions with the NRC in this area have taken place and will continue in an attempt to resolve this issue.

Category IV-C

Item	Branch	Applicable SRP Section	Title
13 6/1/76	CSB	6.2.1.2	Asymmetric Loads on Components Located Within Containment Subcompartments.

RESPONSE

Westinghouse addresses asymmetric loads on components located within containment subcompartments (in compliance with SRP 6.2.1.2) in the Applicants Safety Analysis Reports, since the loads are plant specific.

Category IV-C

Item	Branch	Applicable SRP Section	Title
15 1/1/77	SRP	6.2.1.4	Containment Response Due to Main Steamline Break and Failure of MSLIV to Close

RESPONSE

Westinghouse cannot provide a containment response analysis, since containment design is not in Westinghouse scope. Steamline break mass-energy release source term data can be supplied when required.

Category IV-C

Item	Branch	Applicable SRP Section	Title
17 1/1/77	ASB	9.2.2	Design Requirements for Cooling Water to Reactor Coolant Pumps

RESPONSE

Component cooling water system design will be discussed in Section 9.2.2 of the applicant's PSAR. The portions of the component cooling water system supplying safeguards equipment will be designed to meet single active failure criteria or moderate energy line crack criteria. Abnormal conditions affecting the portions of the component cooling water system which supply non-safeguards equipment (i.e. reactor coolant pumps) will be mitigated by the appropriate.

As discussed in Section 5.5.1, component cooling water is supplied to the pump motor bearing oil coolers and to the pump thermal barrier heat exchanger. Should a loss of component cooling water to the reactor coolant pumps occur, the Chemical and Volume Control System continues to provide injection water to the reactor coolant pumps; the seal injection flow is sufficient to prevent damage to the seals with a loss of thermal barrier cooling. The loss of component cooling water to the motor bearing oil coolers will result in an increase in lube oil temperature and a corresponding rise in bearing metal temperature. Actual testing has shown that the manufacturer's recommended maximum bearing operating temperature will be reached in approximately ten minutes. Therefore, the reactor coolant pumps will incur no damage with a component cooling water flow interruption of 10 minutes.

Westinghouse has conducted a human engineering analysis of this event during normal operation, considering the following factors:

1. The CCW flow indicator and alarm for the CCWS return line from each RCP oil cooler are located on the main control board.

2. The CCW temperature indicator and alarm for the CCWS return line from each RCP oil cooler are located on the main control board.

3. The reactor coolant pump motor bearing temperature s supplied as input to the process computer. A high temperature will cause the computer to alarm and identifv the high temperature.

4. The CCWS isolation valve monitor lights, which would indicate valve closure, are located on the main control board.

5. The psychological stress duced on the average trained operator is much less than that induced by a LOCA (reference WASH-1400) which would cause a response time delay of one minute.

6. The response time required of the operator to trip the reactor and stop the reactor coolant pumps is not complicated and is a direct logical result of the event symptoms as alarmed and inciated.

In conclusion, Westinghouse believes that the NRC'a alternatives given in Item IV.C.17 constitute a significant change in regulatory requirements. Because this issue has not been considered by the RRRC and because a value-impact statement has not been performed, the NRC's request for a change in design is inappropriate.

Category IV-C

Item	Branch	Applicable SRP Section	Title
18 8/1/76	ASB	10.4.7	Design Guidelines for Water Hammer in Steam Generators with Top Feedring Design (BTP ASB-10.2)

RESPONSE

The steam generators employed in the RESAR-41 design are of the preheater type and do not employ top feedring design. Therefore, BTP-ASB-10.2 is not applicable. For a discussion of water hammer in RESAR-41 steam generators, refer to the response to item IV.B.20, above, "Water Hammer for Steam Generators with Preheaters (SRP 10.4.7 par. I.2.b)"

REFERENCES

1. "Westinghouse Water Reactor Divisions Quality Assurance Plan," WCAP-8370, Revision 8A, September 1977.

2. "Nuclear Fuel Division Reliability and Quality Assurance Plan," WCAP-7800, Rev. 5, December 6, 1977.

3. "Methodology for Qualifying Westinghouse PWR-SD Supplied NSSS Safety Related Electrical Equipment, WCAP-8587-R1, September 1977, Supplement 1, Rev. 1, November 1978.

4. "General Method of Developing Multifrequency Biaxial Test Inputs for Bistables," WCAP-8695, September 1975.

5. "Fuel Densification Experimental Results and Model for Reactor Application," WCAP-8218-P-A (Proprietary Class 2), WCAP-8219-A (Non-Proprietary, Class 3), March 1975.

6. "Design, Inspection, Operation, and Maintenance Aspects of the Westinghouse Nuclear Steam Supply System to Maintain Occupational Radiation Exposures as Low as Reasonably Achievable," WCAP-8872, April 1977.

7. Letter of Comment on Rev. 0 of Reg. Guide 1.99 to the Secretary of the Commission by C. E. Eicheldinger, NS-CE-784, September 22, 1975.

8. Hawthorne, J. R., "Radiation Effects Information Generated on the ASTM Reference Correlation - Monitor Steels," to be published.

9. Westinghouse RESAR-3S, Chapter 16, Figure B3/4 4.2, Page B3/4.

10. Letter NS-TMA-1843 to the Secretary of the Commission by T. M. Anderson, June 23, 1978.

11. "Reactor Coolant Pump Integrity in LOCA," WCAP-8163, September 1973.

12. "High Pressure Water Hammer Test of the Split Flow Preheat Steam Generator," WCAP-9232, January 1978.

13. "Damping Valves of Nuclear Power Plant Components," WCAP-7921-AR, May 1974.

14. Marasco, F. W. and Siroky, R. M., "Westinghouse 7300 Series Process Control System Noise Tests," WCAP-8892-A, June, 1977.

15. Letter dated April 20, 1977 from R. L. Tedesco (NRC) to C. Eicheldinger (Westinghouse).

TABLE 3B-1

CROSS REFERENCE OF RESAR-41 TO STANDARD REVIEW PLAN

SRP ACCEPTANCE CRITERIA

RESAR-41 Section	SRP 3.7.3.1	3.7.3.2	3.7.3.3	3.,.3.4	3.7.3.5*	3.7.3.6	3.7.3.7	3.7.3.8	3.7.3.9	3.7.3.10	3.7.3.11	3.7.3.12	3.7.3.13	3.7.3.14	3.7.3.15
3.7.3.1		X													
3.7.3.2				X											
3.7.3.3															
3.7.3.4							X								
3.7.3.5															
3.7.3.6								X							
3.7.3.7						X									
3.7.3.8															
3.7.3.9															
3.7.3.10															
3.7.3.11											X				
3.7.3.12												X			
3.7.3.13													X		
3.7.3.14															
3.7.3.15														X	
3.7.2.1	X		X												
3.7.2.7									X						
3.7.2.9										X					
3.7.2.14															X

* See response to IV.B.11.

TABLE 3B-2

CROSS REFERENCE OF RESAR-41 TO STANDARD REVIEW PLAN

SRP ACCEPTANCE CRITERIA

RESAR-41 Section	SRP 3.7.2.1	3.7.2.2	3.7.2.3	3.7.2.4	3.7.2.5	3.7.2.6	3.7.2.7	3.7.2.8*	3.7.2.9	3.7.2.10	3.7.2.11	3.7.2.12	3.7.2.13	3.7.2.14	3.7.2.15
3.7.2.1	X														
3.7.2.2		X													
3.7.2.3			X												
3.7.2.4				X											
3.7.2.5				X											
3.7.2.6					X										
3.7.2.7															
3.7.2.8									X						
3.7.2.9										X					
3.7.2.10											X				
3.7.2.11												X			
3.7.2.12													X		
3.7.2.13														X	
3.7.2.14															X
3.7.3.4							X								
3.7.3.7						X									

* Not in Westinghouse scope.

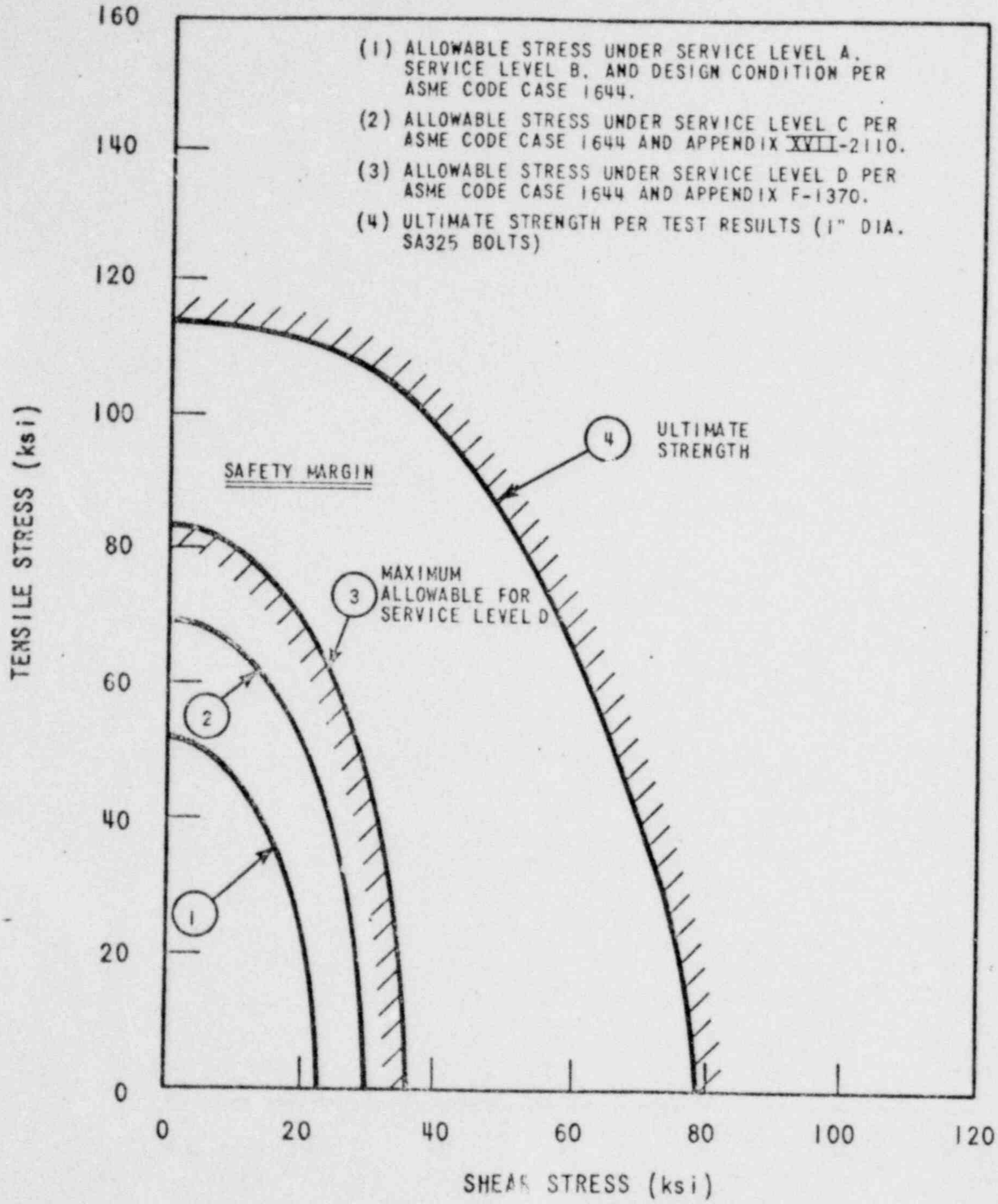

Figure 3B-1 Comparison of Tensile Stress for Bolts

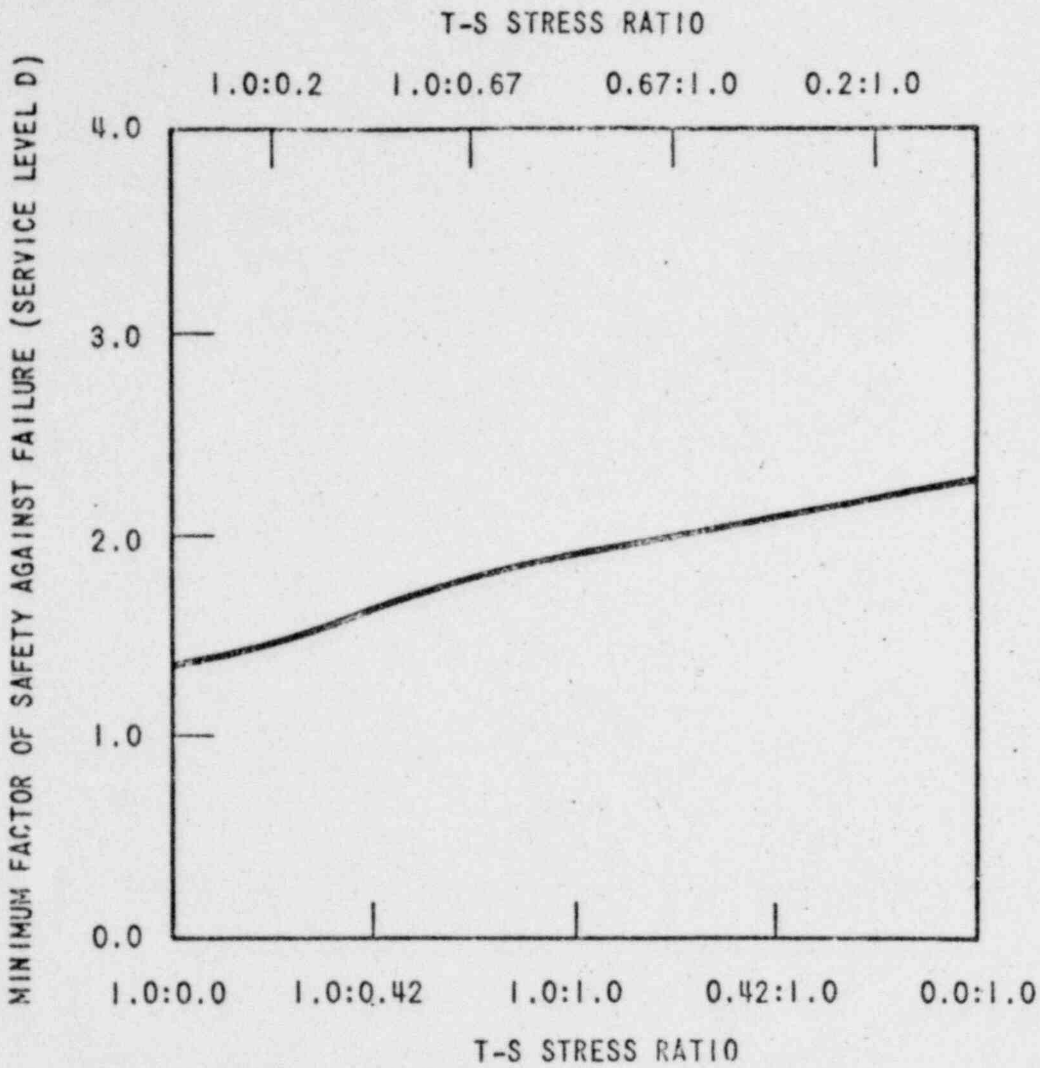

Figure 3B-2 Factor of Safety Againist Failure Under Service Level D as a Function of T-S Ratio