JUL 6 1981

Docket Nos.: STN 50-482
and STN 50-483

APPLICANTS: Union Electric Company
Kansas Gas and Electric Company

FACILITIES: Callaway Plant, Unit 1
Wolf Creek Generating Station, Unit 1

SUBJECT: SUMMARY OF MEETING HELD JUNE 9 AND 10, 1981 WITH CALLAWAY AND WOLF CREEK APPLICANTS TO REVIEW MECHANICAL ENGINEERING

A meeting was held on June 9 and 10, 1981 at the Bechtel offices in Gaithersburg, Maryland with representatives of the Union Electric Company, Kansas Gas and Electric Company, SNUPPS organization, Bechtel Power Corporation and Westinghouse Electric Corporation. This meeting was held as a result of our letter of April 22, 1981 to the applicants transmitting a draft SER and draft questions on the SNUPPS FSAR, and requesting a meeting to discuss those questions. The agenda for the meeting is attached as enclosure 1. The applicants' responses to the draft questions, as revised and agreed to by NRC, are attached as enclosure 3.

The status of the agenda items (enclosure 1) follows.

Topic III.

(a) Bechtel/Westinghouse Division of Responsibility

The applicants indicated that Westinghouse is responsible for all Class 1 piping stress analysis and Bechtel for all Class 2 and 3 piping stress analysis. Westinghouse is responsible for design of the piping supports for the reactor coolant loop and pressurizer surge line, and Bechtel is responsible for design of the supports for Class 1 auxiliary lines.

(b) Comparison to ASB and MEB Criteria

The applicants indicated that in general they are adhering to NRC Branch Technical Posiitions 3-1. In the case of MEB 3-1, paragraph B.3.b(2), the applicant stated that Class 1 auxiliary lines are being fabricated as seamless piping, and that a longitudinal rupture is so low in probability it should be considered incredible. NRC indicated that further review of this position would be required and the issue will remain open. In the case of ASB 3-1, paragraph B.3.b(3), the applicants stated the Auxiliary Feedwater System will be classified the same as a moderate-energy system because it will only be used under accident conditions. A separate startup/shutdown feedwater train is being added to the main feedwater system. This was found acceptable by NRC.

OFFICE ▶							
SURNAME ▶	8107230288 810706 PDR ADOCK 05000482 A PDR			REGULATORY DOCKET FILE COPY			
DATE ▶							

(c) High Energy Line Break Analysis

The applicants indicated to ASB that they would clarify the FSAR to indicate that only the charging systems piping is not considered to have a high energy line break in one train concurrent with a single active failure in a different train. We indicated we would consider it further but did not think we would find it to be a problem since no serious transient could result and it is consistent with previously acceptable plant reviews.

Topic IV. Auxiliary Systems Branch Questions

We questioned whether a pipe failure in a non-safety grade system routed above the Essential Service Water piping might erode the support around the ESW piping. The applicants stated that the ESW piping was embedded in concrete so this would not be a problem. They will revise the FSAR to clarify this.

Topic V. Optional Discussion Items

We toured the SNUPPS scale plant model both days to evaluate selected piping runs, supports, and restraints. An impromptu audit was performed on several piping support calculations. The calculations were found acceptale by the NRC.

Topic VI. MEB SER Draft Questions

There were 31 agenda items discussed which had been identified from the draft questions in the April 22 letter. In addition, 8 additional questions were developed during the meeting, for a total of 39 items. The status of each of these items is listed below.

Item	Status
1. Pg. 3.2 - 2	Resolved by revising FSAR page 3.2-2 as agreed to in the meeting.
2. Pg. 3.6 - 3 (section 3.6.1.1.h.2(b))	Resolved. No change to FSAR required.
3. Pg. 3.6 - 4 (section 3.6.1.1.j)	Resolved. FSAR will be revised per words agreed to in meeting.
4. Section 3.6.1.1.k	Resolved, pending review of an FSAR revision to be provided in July 1981 which will include a summary of results from the Pipebreak Hazards Protection Analysis.

OFFICE ▶							
SURNAME ▶							
DATE ▶							

5. Figure 3.6 -1	Resolved, pending review of FSAR revision to be provided in July 1981 updating Figure 3.6 -1 and providing results of the pipebreak hazards analysis.
6. Section 3.6.2	FSAR will be revised to include a statement that leakage cracks in non-Seismic Category I piping are addressed at all worse case locations.
7. Table 3.6 - 3 (Sheets 1 thru 8)	Resolved. No action required.
8. Table 3.6 - 3	Resolved. FSAR will be revised in July, 1981 to incorporate updated sheets 28, 32, 36.
9. Table 3.6 - 4	Resolved. FSAR will be revised in July, 1981 to reduce the number of "under review" items.
10. General (Protective Measures for Jet Impingement)	Clarified (Resolved). No action required.
11. Pg. 3.7(B) - 7	Resolved. FSAR will be revised as agreed to in meeting to indicate use of Reg. Guide 1.92 Equation 4. Also, reference to sections 5.1 and 5.2 of BP-TOP-1 will be eliminated.
12. Figures 3.7(B) - 5 thru 3.7(B) - 8	Resolved. SAR will be revised, using words agreed to in the meeting, to delete these figures and instead refer to NRC approved topical report BC-TOP-4-A, Rev. 3.
13. Figures 3.7(B) - 9A and - 9D	Resolved. No action required. The seismology area is currently under review by HGEB at at NRC.
14. Pg. 3.7(N) - 14	Resolved per the clarification agreed to in the meeting. FSAR will be revised to correct Equation 3.7(N) - 30.
15. Pg. 3.9(B) - 1 (Section 3.9(B).1.1)	Resolved. FSAR will be revised to clarify whether thermal transients or other dynamic events are referred to.

OFFICE							
SURNAME							
DATE							

16.	General (Thermal shock to RPV internals from ECCS injection after LOCA)	Resolved. FSAR will be revised (section 3.9.5) to incorporate the words agreed to in the meeting. Thermal shock is analyzed under LOCA/ECCS injection conditions.
17.	Pg. 3.9(B) - 1 (Section 3.9(B).1.2.1.1)	Resolved. FSAR will be revised using words agreed to in the meeting. RELAP4 should be referenced in SEction 3.9(B).1.
18.	Pg. 3.9(B) - 3 (Section 3.9(B).1.3.2)	Resolved. FSAR will be revised (pg. 3.9(B)-3) to eliminate "inelastic method or."
19.	Pg. 3.9(B) - 4 (Section 3.9(B).2.1)	Resolved. FSAR will be revised per the meeting response.
20.	Pg. 3.9(B) - 5 (Section 3.9(B).2.1)	Resolved. No action required.
21.	General (Assure the functional capability of Class 1,2, 3 piping essential to safety under all loads)	Open. NRC is considering further. The applicants are preparing to modify their response based on input from Westinghouse. Small piping needs to be addressed in more detail.
22.	Pg. 3.9(B) - 15 (Section 3.9(B).3.3.1.g)	Resolved. FSAR will be revised using words agreed to in the meeting to note there are no instances when a dynamic load factor 2.0 was used.
23.	Pg. 3.9(N) - 33 (Section 3.9(N).2.1	Resolved. FSAR will be revised using words agreed to in the meeting.
24.	Pg. 3.9(N) - 36 (Section 3.9(N) - 2.4)	Resolved, pending review of FSAR revision to summarize the basis for SNUPPS plants being classified as non-prototypic Category I in accordance with Reg. Guide 1.20. Wording used in Commanche Peak FSAR will be considered and is expected to be acceptable.
25.	Table 3.9(N) - 3	Resolved. FSAR will be revised to add the words agreed to in the meeting.
26.	Pg. 3.6 - 10 (Section 3.6.2.1.1.9.2(B))	Resolved. Applicants will revise the FSAR to indicate that 2.4 Sm was used.

27.	Pg. 3.6 - 13 (Section 3.6.2.1.1.e)	Resolved. Welded attachments are not used on high-energy piping in containment penetration areas. Applicant committed to provide location details and advise NRC if welded attachments are used in the future.
28.	Tables 3.9(N) - 2 and - 4	Resolved, pending review of FSAR revision per the meeting response.
29.	Section 3.9.6	Open. A separate submittal will be made by applicant in July 1981 to respond to this question .
30.	General (Submittal of preservice and inservice test program for pumps and valves)	Open. Test program to be submitted by applicant in July 1981.
31.	Pre-Oper. Testing of Snubbers (also Pre-service Inspec.)	Resolved. Applicant provided response in meeting minutes.
32.	RG. 1.121	Resolved. Not an SER item: However, must be reviewed for license tech. specs. before granting an operating license.
33.	Exception to MEB 3-1 (No jet impingement effects are considered for Class I longitudinal breaks)	Open. NRC does not accept the response and feels that further justification is required. The applicant believes NRC should further consider their position but will provide additional justification.
34.	RG. 1.124 and 1.130	Resolved. FSAR will be revised per the words agreed to in the meeting.
35.	Section 3.9(N).3.3.A	Resolved. FSAR will be revised per the applicants' response in the meeting.
36.	Table 3.9(B) - 7	Resolved. FSAR will be revised using words agreed to in the meeting.
37.	Table 3.9(B) - 3 and - 5	Resolved per the discussion in the meeting and the applicants' response.

OFFICE ▶							
SURNAME ▶							
DATE ▶							

38. Pg. 3.9(N) - 44 Resolved by revising the FSAR using the words agreed to in the meeting response.

39. Section 3.9 Resolved by FSAR revision using words agreed to in the meeting response.

Original signed by
A. W. Dromerick

A. W. Dromerick, Project Manager
Licensing Branch No. 1
Division of Licensing

Original signed by
Gordon E. Edison

G. E. Edison, Project Manager
Licensing Branch No. 1
Division of Licensing

Enclosures:
As stated

cc: See next page

OFFICE	DL:LB#1	DL:LB#1	DL:LB#1				
SURNAME	GEdison./ys	ADromerick	BJYoungblood				
DATE	7/2/81	7/ /81	7/ /81				

MEETING SUMMARY DISTRIBUTION

JUL 6 1981

Docket File
NRC PDR
Local PDR
TIC/NSIC/Tera
N. Hughes
LB#1 Reading
H. Denton
E. Case
D. Eisenhut
R. Purple
B. J. Youngblood
A. Schwencer
F. Miraglia
J. Miller
G. Lainas
R. Vollmer
J. P. Knight
R. Bosnak
F. Schauer
R. E. Jackson
Project Manager Edison/Dromerick
Attorney, OELD
M. Rushbrook
OIE (3)
ACRS (16)
R. Tedesco

NRC Participants:

D. Terao (3)
R. Bosnak

G. Lear
WJohnston
S. Pawlicki
V. Benaroya
Z. Rosztoczy
W. Haass
D. Muller
R. Ballard
W. Regan
D. Ross
P. Check
Chief, Power Systems Branch
O. Parr
F. Rosa
W. Butler
W. Kreger
R. Houston
Chief, Radiological Assessment Branch
L. Rubenstein
T. Speis
MSrinivasan
J. Stolz
S. Hanauer
W. Gammill
T. Murley
F. Schroeder
D. Skovholt
M. Ernst
R. Baer
C. Berlinger
K. Kniel
G. Knighton
A. Thadani
D. Tondi
J. Kramer
D. Vassallo
P. Collins
D. Ziemann

bcc: Applicant & Service List

Mr. J. K. Bryan
Vice President - Nuclear
Union Electric Company
P. O. Box 149
St. Louis, Missouri 63166

cc: Gerald Charnoff, Esq.
Shaw, Pittman, Potts,
Trowbridge & Madden
1800 M Street, N. W.
Washington, D. C. 20036

Kansas City Power & Light Company
ATTN: Mr. D. T. McPhee
Vice President - Production
1330 Baltimore Avenue
Kansas City, Missouri 64141

Mr. Nicholas A. Petrick
Executive Director, SNUPPS
5 Choke Cherry Road
Rockville, Maryland 20850

Mr. J. E. Birk
Assistant to the General Counsel
Union Electric Company
St. Louis, Missouri 63166

Kansans for Sensible Energy
P. O. Box 3192
Wichita, Kansas 67201

Mr. Tom Vandel
Resident Inspector/Wolf Creek NPS
c/o USNRC
P. O. Box 1407
Emporia, Kansas 66801

Mr. Michael C. Keener
Wolf Creek Project Director
State Corporation Commission
State of Kansas
Fourth Floor, State Office Building
Topeka, Kansas 66612

Mr. Glenn L. Koester
Vice President - Nuclear
Kansas Gas and Electric Company
201 North Market Street
P. O. Box 208
Wichita, Kansas 67201

Dr. Vern Starks
Route 1, Box 863
Ketchikan, Alaska 99901

Mr. William Hansen
U. S. Nuclear Regulatory Commission
Resident Inspectors Office
RR #1
Steedman, Missouri 65077

Ms. Treva Hearn, Assistant General Counsel
Missouri Public Service Commission
P. O. Box 360
Jefferson City, Missouri 65102

Jay Silberg, Esquire
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Mr. D. F. Schnell
Manager - Nuclear Engineering
Union Electric Company
P. O. Box 149
St. Louis, Missouri 63166

Ms. Mary Ellen Salava
Route 1, Box 56
Burlington, Kansas 66839

Eric A. Eisen, Esq.
Birch, Horton, Bittner & Monroe
1140 Connecticut Avenue, N. W.
Washington, D. C. 20036

Ms. Wanda Christy
515 N. 1st Street
Burlington, Kansas 66839

2/6

ENCLOSURE 1

NRC - SNUPPS MEETING
June 9-12, 1981
AGENDA

I.	SNUPPS Introduction	R.L. Stright
II.	NRC Introduction	R.J. Bosnak
III.	Bechtel/Westinghouse Division of Responsibility Comparison to ASB and MEB Criteria High Energy Line Break Analysis	C.M. Herbst
IV.	Auxiliary Systems Branch Questions	W.T. LeFave
V.	Optional Discussion Items a.stress analysis summary b.Class I analysis c.restraint design d.model tour	
VI.	MEB SER Open Items	(see list attached)
VII.	Summary and Conclusions	R.L. Stright

MEB SER REVIEW MEETING AGENDA ITEMS

Item	Reference	Summary	Response	comments/notes
#1	Page 3.2-2	Non safety-related items that must retain structural integrity	C. Herbst	
#2	3.6.1.1.h.2(b)	Failures of seismic and non-seismic piping	C. Herbst	
#3	3.6.1.1.j	Pipe Whip effects	C. Herbst	
#4	3.6.1.1.k	Line restrictions in pipe break analysis	C. Herbst	
#5	Fig. 3.6-1	Pipe break analysis figures	N. Kalyanam	
#6	3.6.2	Breaks in non-seismic Category I piping	C. Herbst	
#7	Table 3.6-3	Calrification of FSAR Table	N. Kalyanam	
#8	Table 3.6-3	Update FSAR Table	N. Kalyanam	
#9	Table 3.6-4	Update FSAR Table	C. Herbst	

Item	Reference	Summary	Response	comments/notes
#10	3.6	Jet impingement effects on instrumentation	C. Herbst	
#11	Page 3.7(B)-7	Use of BP-TOP-1	N. Kalyanam	
#12	Fig. 3.7(B)-5 thru 3.7(B)-8	Conservatism of response spectra	E. Thomas	
#13	Fig. 3.7(B)-9A and 3.7(B)-9D	Conservatism of response spectra	E. Thomas	
#14	Page 3.7(N)-14	Regulatory Guide 1.92	B. Maurer	
#15	3.9(B).1.1	Transients considered in design of BOP components	N. Kalyanam	
#16	3.9(N)	Thermal shock of RPV internals	S. Boyle	
#17	3.0(B).1.2.1.1	Verification of computer program	N. Kalyanam	
#18	3.9(B)1.3.2	Inelastic methods in stress analysis	N. Kalyanam	

MEB SER REVIEW MEETING AGENDA ITEMS

5 of 6

Item	Reference	Summary	Response	comments/notes
#19	3.9(B).2.1	Thermal expansion and dynamic effects testing	D. Egan A. Passwater G. Rathbun	
#20	3.9(B).2.1	Thermal expansion and dynamic effects testing	C. Herbst	
#21	3.9	Functional capability of ASME Class 1, 2 and 3 piping systems	B. Maurer N. Kalyanam	
#22	3.9(B).3.3.1.g	Dynamic load factor less than 2.0	N. Kalyanam	
#23	3.9(N).2.1	Vibration and dynamic effects testing	B. Maurer	
#24	3.9(N).2.4	Regulatory Guide 1.20	S. Boyle	
#25	3.9(N).3	Specific paragraphs of ASME Section III	B. Maurer	
#26	3.6.2.1.1.9	Pipe break criteria	N. Kalyanam	
#27	3.6.2.1.1.e	Welded attachments on piping in containment penetration areas	N. Kalyanam	

Item	Reference	Summary	Response	comments/notes
#28	Tables 3.9(N)-2 and 3.9(N)-4	Load combination	B. Maurer	
#29	3.9.6	Isolation of RCS from low pressure systems	C. Herbst C. Hultman	
#30	3.9	Pump and valve test program	C. Hultman	
#31	3.9	Pre-service examination and pre-operational testing of snubbers	H. Borda C. Hultman	

ENCLOSURE 2

NRC Meeting at Bechtel Offices
in Gaithersburg, MD. on SNUPPS FSAR
For Callaway Unit 1 and Wolf Creek
Unit 1 Nuclear Plants
June 9, 1981

	NAME	ORGANIZATION
1.	R. L. Stright	SNUPPS STAFF
2.	J. M. Small	BECHTEL
3.	P. A. Ward	BECHTEL
4.	John S. Prebula	BECHTEL
5.	Kathy Miller	BECHTEL
6.	William Poppe	BECHTEL
7.	Charles Herbst	BECHTEL
8.	N. P. Goel	BECHTEL
9.	John Hurd	BECHTEL
10.	Bhupesh Shah	BECHTEL
11.	N. Kalyanam	BECHTEL
12.	Rena Lee	BECHTEL
13.	Hector E. Borda	BECHTEL
14.	Joseph H. Smith	BECHTEL
15.	Jim Alzheimer	PNL for NRC
16.	Godon Beeman	PNL/NRC
17.	David Terao	NRC/MEB
18.	H. L. Brammer	NRC/MEB
19.	Bob Bosnak	NRC/MEB
20.	G. E. Edison	NRC/DL
21.	Y. L. Li	NRC/MEB
22.	R. A. Jaross	ANL/NRC
23.	W. T. Le Fave	ASB/NRC
24.	A. C. Passwater	UE
25.	G. P. Rathbun	KG&E
26.	C. W. Hultman	SNUPPS
27.	D. W. Capone	UE
	PART-TIME	
28.	B. L. Meyers	BECHTEL
29.	M. Stuchfield	BECHTEL
30.	W. L. Luce	WESTINGHOUSE
31.	J. J. Mc Inerney	WESTINGHOUSE
32.	B. Maurer	WESTINGHOUSE
33.	E. Thomas	BECHTEL
34.	K. Lee	BECHTEL
35.	N. Singleton	WESTINGHOUSE
36.	Deo Ray Bliandum	WESTINGHOUSE
37.	R. W. Beer	WESTINGHOUSE
38.	L. S. Shockling	WESTINGHOUSE

SNUPPS MEETING ATTENDEES
June 10, 1981

	NAME	ORGANIZATION
1.	R. L. Stright	SNUPPS STAFF
2.	P. A. Ward	BECHTEL
3.	J. S. Prebula	BECHTEL
4.	W. A. Poppe	BECHTEL
5.	G. E. Edison	NRC/DL
6.	K. A. Miller	BECHTEL
7.	C. M. Herbst	BECHTEL
8.	W. P. GOEL	BECHTEL
9.	Bhupesh Shah	BECHTEL
10.	N. Kalyanam	BECHTEL
11.	Rena Lee	BECHTEL
12.	H. E. Borda	BECHTEL
13.	J. H. Smith	BECHTEL
14.	Jim Alzheimer	PNL for NRC
15.	Gordon Beeman	PNL/NRC
16.	David Terao	NRC/MEB
17.	H. L. Brammer	NRC/MEB
18.	R. J. Bosnak	NRC/MEB
19.	Y. L. Li	NRC/MEB
20.	F. C. Cherny	NRC/MEB
21.	B. F. Maurer	W-SMD
22.	W. L. Luce	W-Licensing
23.	J. J. Mc Inerney	W-Licensing
24.	A. C. Passwater	UE
25.	G. P. Rathbun	KG&E
26.	C. W. Hultman	SNUPPS
27.	B. L. Meyers	BECHTEL
28.	D. W. Capone	UE

ENCLOSURE 3

Callaway and Wolf Creek Plants

Applicants' Responses to NRC

Draft Questions as Agreed with

NRC (MEB) at Review Meetings

June 9 and 10, 1981

#1. Page 3.2-2

"Nonsafety-related structures, systems, and components that must be designed to retain structural integrity during and after an SSE, but do not have a function, are seismically analyzed." Assurance should be made that the above items meet the faulted limits. It is also stated that these above items are not controlled by a 10 CFR 50 Appendix B Quality Assurance Program. These items should be included in the Quality Assurance Program or a program providing an equivalent level of quality assurance.

RESPONSE

See revised Page 3.2-2.

::

All components classified as Safety Class 1, 2, or 3 (classifications are as defined by Reference 1), are seismic Category I.

Seismic Category I structures, components, and systems are designed to withstand the safe shutdown earthquake (SSE), as discussed in Sections 3.7(B) and 3.7(N), and other applicable load combinations, as discussed in Sections 3.8.1 through 3.8.5. Seismic Category I structures are sufficiently isolated or protected from the other structures to ensure that their integrity is maintained.

Radwaste systems and structures are designated as nonseismic Category I. In accordance with Regulatory Guide 1.143, a simplified seismic analysis is performed for portions of the gaseous radwaste system (which by design are intended to store and delay the release of gaseous radioactive waste), including isolation valves, equipment, interconnecting piping, and components located between the upstream and downstream valves used to isolate these components from the rest of the system. In addition, a simplified seismic analysis is performed for structures housing radioactive waste management systems in accordance with Regulatory Guide 1.143.

Nonsafety-related structures, systems, and components that must be designed to retain structural integrity during and after an SSE, but do not have to function, are seismically analyzed to ensure that faulted stress limits are not exceeded. These items (for example: piping and piping supports for nonsafety-related piping located over safety-related items) whose continued function is not required are nonseismic Category I and are not controlled by a 10 CFR 50 Appendix B Quality Assurance Program (not Q-listed). The nonseismic Category I Systems Quality Assurance Program is described in Section 17.D of the SNUPPS Quality Assurance Programs for Design and Construction.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATION

The quality group classification for each water- and steam-containing pressure component is shown in Table 3.2-1. The components are classified according to their importance to safety, as dictated by service and functional requirements and by the consequences of their failure. The quality group classifications and code requirements for the quality of plant process systems meet the intent of Regulatory Guides 1.26 and 1.143. Clarifications and specific exceptions to these guides are discussed in Tables 3.2-4 and 3.2-5, respectively. These tables compare the design to each regulatory position.

The design, fabrication, inspection, and testing requirements of each classification provide the required degree of conservatism in assuring component pressure integrity and operability.

#2. Section 3.6.1.1.h.2(b), Page 3.6-3

It is stated that it was assumed the failure of seismic Category I and seismically supported nonseismic Category I piping was caused by some mechanism other than an earthquake and, therefore, that nonseismic Category I equipment could be used to bring the plant to a safe shutdown. What mechanisms are postulated for failure of seismic Category I and seismically supported nonseismic Category I piping? Assurance must be made that the failed seismic piping does not damage the nonseismic Category I equipment mentioned above. Assurance must also be made that only seismic Category I equipment will be used to bring the plant to a safe shutdown in the event of an SSE.

RESPONSE

Seismic Category I and seismically supported nonseismic Category I piping systems are assumed to fail nonmechanistically for the purpose of pipe break hazards analysis.

~~In some cases.~~ Nonseismic Category I equipment could [redacted] be utilized to bring the plant to safe shutdown following a postulated pipebreak event, since a seismic event is not assumed to occur simultaneously with a pipebreak.

As stated in Section 3.6.1.1.h.2(b), only seismic Category I equipment is assumed to be available to bring the plant to a safe shutdown following an SSE.

#3. Section 3.6.1.1.j, Page 3.6-4

It is stated that the pipe whip was assumed to occur in the plane defined by the piping geometry and to cause movement in the direction of the jet reaction. Assurances must be made that this criteria was used only in the design of pipe whip restraints and that failed piping was considered capable of swinging in any direction about a plastic hinge following a pipe rupture and all potential targets were considered.

RESPONSE

Jet impingement targets are identified in accordance with Standard Review Plan 3.6.2, based on the evaluated movement of the pipe. Pipe whip restraints are provided wherever postulated pipe breaks have any possibility of affecting any system or component required for the mitigation of that break or safe shutdown of the plant. Unrestrained pipe breaks are limited only to those areas of the plant that are physically separated from the systems and components required for pipe break mitigation or safe shutdown. In general, whipping ends from a pipebreak are restrained, such that plastic hinge formation is not allowed to occur. Where equipment, piping systems, raceways, etc. were considered to be the targets of pipebreak fluids, an evaluation of the hazard is performed on an individual case basis. FSAR section 3.6 will be modified to include the second and third sentences of the above response.

#4. Section 3.6.1.1.k

All instances where line restrictions or the absence of energy reservoirs were used in the calculation of thrust and jet impingement forces should be listed.

RESPONSE

The analysis of the effects of each pipebreak event is described in the Pipebreak Hazards Protection Analysis. Line restrictions and limited reservoirs have been considered in such cases where they exist. A summary of the results of this analysis is being included in an FSAR revision which will be submitted in July 1981.

#5. Figure 3.6-1

Various sheets indicate that pipe break restraint locations, Class 1 analysis pipe break locations, and effects analysis for high-energy pipe breaks located within containment are all under review. We cannot complete our review until these reviews are completed.

RESPONSE

Figure 3.6-1 was included in the initial submittal of the SNUPPS FSAR in October 1979 to report on the results of piping systems stress analysis. Since then, the stress analyses have been updated to include changes in the input information that resulted from refinements in the design. In addition, the results of some analyses that were not complete at the time of the FSAR submittal have since been completed. All of these updates will be reported in revised Figure 3.6-1.

The Pipebreak Hazards Protection Analysis program will provide revised pipebreak locations and pipebreak restraint locations. The Pipebreak Hazards Protection Analysis program also provides high-energy pipebreak effect analysis. The results of this program are scheduled to be submitted in July 1981 in the form of an FSAR revision.

#6. Section 3.6.2

A statement should be made that breaks and leakage cracks in nonseismic Category I piping are postulated in worse case locations and that failure of non-seismic Category I piping will not cause failure of seismic equipment.

RESPONSE

As stated in FSAR Section 3.6.2.1.1.d, breaks in non-nuclear piping were postulated in each run or branch run at terminal ends of the runs and at all intermediate fittings (e.g., elbows, tees, reducers, welded attachments, and valves), consistent with Standard Review Plan 3.6.2.

As stated in FSAR Section 3.2.1, seismic Category I equipment is protected against the failure of nonseismic Category I equipment. Leakage cracks in nonseismic Category I piping are postulated in worse case locations. FSAR section 3.6 will be modified to include the ~~last~~ preceding sentence.

#7. Table 3.6-3

What is the difference between Sheet 1 and Sheet 2, Sheet 3 and Sheet 4, Sheet 5 and Sheet 6, and Sheet 7 and Sheet 8?

RESPONSE

Sheets 1, 2, 3, 4, 5, 6, 7, and 8 list the stress results for problems 001, 001A, 002, 002A, 003, 003A, 004, and 004A, respectively.

#8. Table 3.6-3

Sheets 28, 32 and 36 indicate that the stress analysis is under review. We cannot complete our review of Section 3.6.2 until this information is furnished.

RESPONSE

Sheets 28, 32, and 36 have been updated to incorporate the latest stress analysis results. The remainder of Table 3.6-3 is being updated to reflect refinements in the piping system stress analysis and will be provided in the form of an FSAR revision in July 1981.

TABLE 3.6-3 (Sheet 28)

SYSTEM - CHEMICAL AND VOLUME CONTROL SYSTEM

Prob. No. P-119
Issue - 2

Node	Stress (psi) Primary	Secondary	Total	Pipe Break Stress Limit (psi) $0.8 (S_A + 1.2S_h)$
45BB	5,349	13,958	19,307	37,244
47M	5,017	19,116	24,133	37,244
49	18,238	15,476	34,395	37,244
60T	15,515	7,989	23,504	37,244
145M	9,021	19,464	28,485	37,244
160M	10,232	21,632	31,864	37,244
220M	4,066	18,211	22,277	37,244
245E	4,057	19,998	24,055	37,244
270*	3,980	3,553	7,533	37,244

* - Indicates Terminal End

TABLE 3.6-3 (Sheet 32)

SYSTEM - CHEMICAL AND VOLUME CONTROL

Prob. No. P-146
Issue - 2

Node	Stress (psi) Primary	Secondary	Total	Pipe Break Stress Limit (psi) 0.8 (S_A + $1.2S_h$)
5*	5,469	5,201	10,670	37,648
30T	7,194	23,858	31,052	37,648
35T	8,722	13,492	22,214	37,648
40T	8,741	10,360	19,101	37,648
44T	10,526	12,356	22,882	37,648
48	11,182	8,537	19,719	37,648
80T	12,578	6,878	19,456	37,648
102T	12,189	22,635	34,824	37,648
106	12,288	10,262	22,550	37,648
130T	15,138	6,132	21,270	37,648
202T	11,671	8,697	20,368	37,648
401*	15,986	22,849	38,835	37,648
315*	7,077	630	7,707	37,648

* - Indicates Terminal End

TABLE 3.6-3 (Sheet 36)

SYSTEM - STEAM GENERATOR BLOWDOWN

Prob. No. P-219
Issue - 1

Node	Stress (psi) Primary	Stress (psi) Secondary	Stress (psi) Total	Pipe Break Stress Limit (psi) = 0.8 (S_A + 1.2S_h)
5	4,755	16,362	21,117	32,400
205	3,085	6,435	9,520	32,400
95	6,559	412	6,971	32,400
170	13,272	9,836	23,108	32,400
175	7,499	5,976	13,475	32,400
178*	17,893	20,834	38,727	32,400
A20	4,491	33,600	38,091	32,400
A30	3,863	16,605	20,468	32,400
B35	8,283	9,143	17,426	32,400
B60	8,162	11,297	19,459	32,400
C35	9,304	14,845	24,149	32,400
A50	4,765	1,877	6,642	32,400
A63	7,333	1,011	8,344	32,400
192T	6,677	28,319	34,996	32,400
240*	3,614	383	3,997	32,400
203	2,783	4,180	6,963	32,400
260	4,818	1,328	6,146	32,400
255	5,480	1,233	6,713	32,400
D15	10,816	5,151	15,967	32,400
E50	7,237	3,362	10,599	32,400
F20	4,414	9,291	13,705	32,400
D15	8,653	5,076	13,729	32,400
F25*	5,868	19,607	25,475	32,400
C40*	12,973	32,498	45,471	32,400

* - Indicates Terminal End

#9. Table 3.6-4

Data in this table under effects analysis are listed as (under review). We cannot complete our review of Section 3.6.2 until this information is furnished.

RESPONSE

High-energy pipebreak effects analysis results are being completed room-by-room as part of the pipebreak analysis. The results of the analysis are being included in an FSAR revision which will be submitted in July 1981.

#10 General

The applicant should describe in detail the protective measures used to mitigate the consequences of jet impingement on safety-related instrumentation inside containment.

RESPONSE:

Protection from the consequences of jet impingement onto safety-related instrumentation inside containment is provided by optimizing instrumentation locations; by optimizing cable and sensing line routing and by use of jet barriers if necessary. All pipebreaks inside containment are restrained in a manner such that uncontrolled jets are prevented and the cone of influence of each jet is limited. Thereby, the hazards of jet impingement are not allowed to adversely affect safety-related instrumentation inside containment.

11. Page 3.7(B)-7

Reference is made to FSAR Section 3.7(B).2.7 which references Sections 5.1 and 5.2 of BP-TOP-1 for the criteria used for combining modal responses for piping systems: The last sentence in Section 5.2 of BP-TOP-1 (page 14) includes the words "if they do occur in-phase" with regard to when the grouping method or the double sum method will be used for closely spaced modes. Please indicate how closely spaced modes were determined to "occur in-phase" and give an example of when they were determined not to occur in-phase.

<u>RESPONSE</u>

For piping systems, closely spaced modes were determined per NRC Regulatory Guide 1.92, Equation 4. FSAR Section 3.7(B).2.7 has been revised to incorporate this statement. Also the sentence, "Sections 5.1 and 5.2 of BP-TOP-1 describe the criteria used for piping systems" has been eliminated from Section 3.7(B).2.7.

Models, typically shown in Figure 3.7(B)-13, were used to perform soil-structure interaction analyses for all four sites. For each site, the site dependent soil properties were used. The vertical dimension of each soil element is equal to or less than Cs/5f, where Cs is the lowest soil element shear wave velocity reached during iterations and f is the highest frequency of interest to be transmitted through the soil profile. The highest frequency used was 25 Hz. In the analyses for the same buildings with site dependent soil parameters, the structural elements remained unchanged.

The site dependent soil properties consisted of strain dependent damping and modulus relationships for each material. In general, the soil properties are nonlinear in character. An iterative process was used to obtain equivalent linear properties which are strain dependent. The methods generally used for such an analysis are included in the computer program FLUSH.

3.7(B).2.5 Development of Floor Response Spectra

Acceleration time-histories obtained from the FLUSH finite element analyses were used in computing the floor response spectra for the major seismic Category I structures. The spectra were generated following the procedures outlined in Section 5.2 of BC-TOP-4-A, using the SPECTRA computer program (see subparagraph 3.8A.12).

3.7(B).2.6 Three Components of Earthquake Motion

Procedures for considering the three components of earthquake motion in determining the seismic response of structures, systems, and components follow the recommendations of Regulatory Guide 1.92 and are described in Section 4.3 of BC-TOP-4-A and Section 5.1 of BP-TOP-1.

3.7(B).2.7 Combination of Modal Responses

Combination is done according to the criterion of "the square-root-of-the-sum-of-the-squares" (SRSS).

Section 4.2.1 of BC-TOP-4-A describes the techniques used to combine modal responses for structures and equipment. For piping systems, closely spaced modes were determined per NRC Regulatory Guide 1.92, Equation 4.

3.7(B).2.7.1 Significant Dynamic Response Modes

The static load equivalent or static analysis method involves the multiplication of the total weight of the equipment or component member by the specified seismic acceleration. Multiple degree-of-freedom systems which may have had frequencies in the resonance region of the amplified response spectra curves

#12. Figures 3.7(B)-5 through 3.7(B)-8

The response spectra of the synthetic time-history does not envelope the corresponding design spectra for all frequencies. Please explain this apparent non-conservatism.

RESPONSE

Figures 3.7(B)-5 through 3.7(B)-8 are consistent with Figures 2-13, 2-14, 2-17, and 2-18 of BC-TOP-4-A, Rev. 3, "Seismic Analysis of Structures and Equipment for Nuclear Power Plants." These figures will be deleted from the FSAR, and reference will be made to BC-TOP-4-A, Rev. 3, when referring to these figures.

as 60 percent of the SSE. The values shown are for the site with maximum amplification. Section 2.5.2 of each Site Addendum and Section 2.5 of BC-TOP-4-A (Ref. 3) discuss the effects of focal and epicentral distances from the site, depths between the focus of the seismic disturbances and the site, existing earthquake records, and the associated amplification of the response spectra.

Earthquake duration influences only the number of loading cycles on equipment because the equipment is designed for the elastic range in accordance with the analytical procedures outlined in BC-TOP-4-A. A 20.48-second duration is considered to be adequate for the time-history type of analysis used for the structures and equipment.

The design response spectra and earthquake time-histories are applied in the free field at finished grade for all sites, except the Tyrone site where the design response spectra and earthquake time-histories are conservatively applied at top of rock below grade. For differences between subsurface conditions at the Tyrone site and those at the other three sites, see Figures 3.7(B)-11A and B.

3.7(B).1.1.1 Bases for Site Dependent Analysis

Section 2.5.2 of each Site Addendum and BC-TOP-4-A, Sections 2.4 and 2.5, describe the bases for specifying the vibratory ground motion for design use.

3.7(B).1.2 Design Time History

Synthetic earthquake time-histories were generated because the response spectra of recorded earthquake motions do not necessarily envelope any of the sites' design spectra. Figures 3.7(B)-3 and 3.7(B)-4 show the synthetic earthquake time-history motions in the horizontal and vertical directions, respectively. The time-histories shown were truncated to 20.48 seconds for use in the FLUSH finite element analyses discussed in Section 3.7(B).2.4.2. Figures 2-13, 2-14, 2-17, and 2-18 of BC-TOP-4-A show that the response spectra of the synthetic time-histories for the horizontal and vertical directions envelope the corresponding design spectra for 1 percent, 2 percent, 5 percent, 7 percent, and 10 percent damping. Section 2.5.1 of BC-TOP-4-A describes the generation of a typical synthetic earthquake time-history.

Typical foundation-level, free-field acceleration response spectra for each of the four sites are presented in Figures 3.7(B)-9A through D. Their envelope is presented in Figure 3.7(B)-10. All curves overlay the SNUPPS 60-percent design response spectra.

Figures 3.7(B)-5 through 3.7(B)-8 are to be deleted from the Standard Plant FSAR.

#13 Figures 3.7(B)-9A and 3.7(B)-9D

Please explain the significance and conservatism of these figures.

RESPONSE

Figures 3.7(B)-9A and 3.7(B)-9D present horizontal SSE free field acceleration response spectra computed at the bottom of the auxiliary / control building for the Callaway and Wolf Creek sites, respectively. In order to demonstrate the conservatism of the seismic input used, the free field spectra are compared in these figures with the SNUPPS 60-percent horizontal design response spectrum. Since the free field spectra for the most deeply embedded power block foundation are considered, the figures represent worst-case comparisons for all the power block structures. The design of all power block structures, systems, and components is based on the envelope of responses for multiple sites. This procedure leads to an enveloping of the 60-percent design spectrum. Consequently, the seismic input used is conservative with respect to the 60-percent design response spectrum criterion in all cases.

#1 Page 3.7(N)-14

Equation [3.7(N)-29] is not necessarily conservative with respect to the requirements of Reg. Guide 1.92. Provide justification for its acceptability. Equation [3.7(N)-30] is in error.

RESPONSE

The method used by Westinghouse for the combination of closely spaced modes has been accepted previously by the NRC (i.e.,RESAR-41, RESAR-414, and numerous plant dockets) as an acceptable alternative to the recommendations of Regulatory Guide 1.92. The Mechanical Engineering Branch will notify the Structural Engineering Branch that, on this basis, the item is considered closed.

Additionally, FSAR Equation 3.7(N)-30 will be revised in a future revision to correct an editorial error.

#15 Section 3.9(B).1.1, Page 3.9(B)-1

Reference is made to section 3.9(N).1.1. Section 3.9(N).1.1 discusses the transients considered in the design of the reactor coolant system (RCS), RCS component supports, and reactor intervals. Are these the same transients used in the design of the BOP components?

RESPONSE

Class 1 branch piping and components are designed and analyzed using the design transients used to analyze the RCS, RCS component supports, and reactor internals as described in Section 3.9(N).1.1.

Class 2 and 3 ~~and non-Section III~~ BOP piping systems and components do not require thermal transient analysis. Class 2 and 3 piping systems and components are designed and analyzed for dynamic transients, ~~including fast valve closure,~~ as identified in Section 3.9(B).2, in accordance with Section III of the ASME Code for normal, upset, and faulted conditions.

Section 3.9(B).1.1 and 3.9(B).2.1.a have been revised accordingly.

3.9(B) MECHANICAL SYSTEMS AND COMPONENTS

3.9(B).1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9(B).1.1 Design Transients

Refer to Section 3.9(N).1.1 for a description of the operating conditions considered in the design of the RCS, RCS component supports, and reactor internals. Class 1 piping systems are designed and analyzed using design transients that are compatible with those described in Section 3.9(N).1.1.

Class 2 and 3 a~~nd non-Section III~~ piping systems and components do not require thermal transient analysis.

Class 2 and 3 piping systems and components are designed and analyzed for dynamic transients ~~including fast valve closure~~, as listed in Section 3.9(B).2.

3.9(B).1.2 Computer Programs Used in Analyses

For NSS systems, refer to Section 3.9(N).1.2.

3.9(B).1.2.1 Seismic Category I Items Other Than the NSSS

Table 3.9(B)-1 lists computer programs used in the balance-of-plant system components. The verification of programs is as follows:

3.9(B).1.2.1.1 ME-632 Program

The ME-632 program is used to determine stresses and loads due to thermal expansion, deadweight, and transient force functions such as those created by fast relief valve opening and closing, pipe break, or fast activation of high-capacity pumps (water hammer effects).

The results obtained from pipe stress program ME-632 have been compared with a) ASME Benchmark problem results, b) Pipe Stress Program TPIPE, c) general purpose program ANSYS, and d) long-hand calculations. The comparison of the results are given in the verification report of the ME-632 program (Ref. 3).

A description of this computer code is included in Table 3.9(B)-1.

Appendix 3.9(B)A provides a verification report for the ME-632 program.

3.9(B).1.2.1.2 ME-101 and TPIPE Programs

The ME-101 and TPIPE computer programs are used to determine stresses and loads due to restrained thermal expansion, deadweight, seismic anchor movement, and earthquake in the following piping:

a. Seismic Category I ASME Section III Class 1, 2, and 3 piping 2 1/2 inches and larger.

time dependent forcing function, such as fast valve closure, while the second is a constant vibration, usually flow induced.

a. Transient response

Dynamic events falling in this category are anticipated operational occurrences. The systems are operated in their normal mode (emergency mode for auxiliary feedwater turbine pump), and measurements are recorded on the systems during and following the event that causes the transient induced vibrations. The systems and the associated transients to be included in the preoperational test program to verify the piping system are:

1. Main steam

   (a) Main steam turbine stop valve trip

   (b) Main steam atmospheric dump valves opening

   (c) Main steam condenser dump valves opening

2. Pressurizer power-operated relief valve blowdown

3. Auxiliary feedwater pump turbine stop valve trip

Selected snubbers subjected to the above transients are monitored during this preoperational testing to assure proper snubber operation.

All of the above are upset transients and a time dependent dynamic analysis is performed on the system. The stresses thus obtained are combined with system stresses resulting from other operating conditions in accordance with the criteria provided in Table 3.9(B)-2.

b. Steady state vibration

System vibration resulting from flow disturbances falls into this category. Positive displacement pumps may cause such flow variation and vibrations and, as such, will be reviewed. Such systems will be checked, including the charging systems.

Since the exact nature of the flow disturbance is not known prior to pump operation, no analysis is performed. A visual steady state vibration inspection

#16 The thermal shock in the RPV internals due to an ECCS injection following a design basis LOCA should be addressed.

RESPONSE

As shown in Table 3.9(N)-1 a LOCA is defined as a faulted design condition. Since a LOCA is accompanied by an ECCS injection, the thermal shock from this injection ~~in~~ is included in the evaluation of the LOCA transient for the reactor internals. Additionally, other upset and emergency condition thermal transients, such as inadvertent safety injection, are included in the evaluation of reactor internals.

Stresses due to thermal shock following an ECCS injection have been evaluated and shown to satisfy the requirements of the ASME Code, Appendix F, as defined in the SNUPPS FSAR. In summary, peak stresses in the reactor internals due to thermal shock do not cause any loss of function.

FSAR Section 3.9.5 will be revised to ~~address~~ state that thermal shock from an ECCS injection following a LOCA for reactor internals has been analyzed.

17 Section 3.9(B).1.2.1.1, Page 3.9(B)-1

This Section references Appendix 3.9(B)A which states that ME-632 results were compared with the results of the previously approved Engineering Data System (EDS) computer programs. Where is a discussion of the verification of the EDS programs and when was it approved?

RESPONSE

FSAR Appendix 3.9(B)A, Page 3.9(B)A-1 has been revised to eliminate "previously approved." Also, Section 3.9(B).1.2.1.1, Page 3.9(B)-1 and Section 3.9(B).7, Page 3.9(B)-20 have been revised. FSAR Section 3.9 B will be revised to reference computer codes used to calculate dynamic forcing functions and time histories and the method of verification.

APPENDIX 3.9(B)A

ME-632 VERIFICATION REPORT

The following is a comparison of the ME-632 program results with the results of the Engineering Data System computer program.

The two piping systems chosen for stress checks were:

a. The Core Spray Piping System - Monticello Nuclear Generating Plant Unit 1

b. Lines 48223-18-HE, 50056-10-HE, and 50057-10-HE-SMUD Rancho Seco Unit 1

These two test cases were chosen because independent piping stress analyses performed by Engineering Data Systems (EDS) under contract to Bechtel were available for comparison purposes. The EDS (PISOL 3) analysis of the core spray piping system consisted of both deadweight and thermal loading while the SMUD Rancho Seco piping system was an earthquake response spectrum analysis.

The ME-632 piping stress analyses were performed in the September 18-20, 1972 period on PICC's Honeywell 635 computer. A relocatable binary deck of the program is stored on tape No. 8312 and will be retained indefinitely for documentation purposes.

A comparison of the ME-632 and EDS analyses is shown in Table 3.9(B)A-1. Due to differing sign conventions, the reactions have opposite signs. The EDS program prints the effects of the support on the piping system while ME-632 prints the effect of the piping system on the support. In some cases, the maximum values for the ME-632 analysis occurred at the middle of the bend. However, since the EDS program does not compute output quantities at the middle of a bend, these maximums are not shown in Table 1. The maximums shown in the table occurred at the same physical point on the piping system in both analyses.

In all cases, the maximum difference in output quantities was less than 5 percent, based upon the corresponding peak value for the particular load case.

It is, therefore, concluded that ME-632 correctly performs static and thermal analysis of piping systems, consistent with the assumptions of the elastic beam theory and applicable flexibility and stress intensification factors specified in ASME Section III.

3.9(B) MECHANICAL SYSTEMS AND COMPONENTS

3.9(B).1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9(B).1.1 Design Transients

Refer to Section 3.9(N).1.1 for a description of the operating conditions considered in the design of the RCS, RCS component supports, and reactor internals. Class 1 piping systems are designed and analyzed using design transients that are compatible with those described in Section 3.9(N).1.1.

Class 2 and 3 and non-Section III piping systems and components do not require transient analysis.

3.9(B).1.2 Computer Programs Used in Analyses

For NSS systems, refer to Section 3.9(N).1.2.

3.9(B).1.2.1 Seismic Category I Items Other Than the NSSS

Table 3.9(B)-1 lists computer programs used in the balance-of-plant system components. The verification of programs is as follows:

3.9(B).1.2.1.1 ME-632 Program

The ME-632 program is used to determine stresses and loads due to thermal expansion, deadweight, and transient force functions such as those created by fast relief valve opening and closing, pipe break, or fast activation of high-capacity pumps (water hammer effects).

The results obtained from pipe stress program ME-632 have been compared with a) ASME Benchmark problem results, b) Pipe Stress Program TPIPE, c) general purpose program ANSYS, and d) long-hand calculations. The comparison of the results are given in the verification report of the ME-632 program (Ref. 3).

A description of this computer code is included in Table 3.9(B)-1.

Appendix 3.9(B)A provides a verification report for the ME-632 program.

3.9(B).1.2.1.2 ME-101 and TPIPE Programs

The ME-101 and TPIPE computer programs are used to determine stresses and loads due to restrained thermal expansion, deadweight, seismic anchor movement, and earthquake in the following piping:

a. Seismic Category I ASME Section III Class 1, 2, and 3 piping 2 1/2 inches and larger.

b. Seismic Category I ASME Section III Class 2 and 3 piping 2 inches and smaller that cannot be analyzed per ME-602.

c. ANSI B31.1 Power Piping Included in High Energy Piping Systems.

3.9(B).7 REFERENCES

1. "Program ME-101 and ME-632 Seismic Analysis of Piping Systems, Users Manual," Pacific International Computing Corp., March, 1971.

2. BP-TOP-1, Seismic Analysis of Piping Systems, Bechtel Power Corporation, San Francisco, California, Rev. 3, January, 1976.

3. "Seismic Analysis of Piping Systems Program ME-632 Verification Report," Version B10, Bechtel Power Corporation.

\# 18 Section 3.9(B).1.3.2, Page 3.9(B)-3

It is indicated that inelastic methods are not used in the design of Code or non-Code components for the faulted condition. On Page 3.9(B)-4, Section 3.9(B).1.4.2 it is indicated that inelastic analyses were used. Please clear up the discrepancy.

RESPONSE

FSAR Section 3.9(B).1.3.2, Page 3.9(B)-3 has been revised to eliminate "inelastic method or."

The program is based on Welding Research Council Bulletin 107, August 1965. The program has been verified based upon hand calculations.

3.9(B).1.2.1.6 CE901 ICES/STRUDL-II

The ICES/STRUDL-II code is used in the design of component supports. For ASME Section III Class 1 piping support design, the program is used to obtain stiffness properties of the support. The results of the analyses are incorporated into overall reactor vessel internal models which calculate the dynamic response due to seismic and LOCA conditions and yield dynamic stresses. In the design of ASME Section III Class 2 and 3 piping supports, models of certain indeterminate support designs are programmed in order to obtain support loads and stresses.

A description and validation of this program are included in Section 3.8A.1.10 of Appendix 3.8A.

3.9(B).1.2.1.7 CE800 (BSAP), CE802 (SPECTRA), and CE786

These programs were used to determine the seismic response spectra of the NSSS for reactor coolant loop branch piping analysis, stresses, and displacements of the main feedwater and main steam system in the reactor building, and to determine seismic anchor movements of the NSSS for incorporation into the piping analysis.

A description and validation of these programs are included in Sections 3.8A.1.5, 3.8A.1.6, and 3.8A.1.8 of Appendix 3.8A.

3.9(B).1.3 <u>Experimental Stress Analysis</u>

3.9(B).1.3.1 NSS System

Refer to Section 3.9(N).1.3.

3.9(B).1.3.2 Seismic Category I Items Other Than the NSSS

Experimental stress analysis methods are not used in the design of Code or non-Code components for the faulted condition. For code components, the stresses will not exceed the limits of the ASME B and PV Code, Section III.

3.9(B).1.4 <u>Considerations for the Evaluation of the Faulted Condition</u>

A listing of all seismic Category I safety-related mechanical systems and components is included in Table 3.2-1.

#9 Section 3.9(B).2.1, Page 3.9(B)-4

More information is needed regarding the piping vibration, thermal expansion and dynamic effects testing programs. Please list those systems to be monitored for 1) transient induced vibration, 2) steady state vibration and 3) thermal expansion. Also list the flow modes of operation to be included in the testing program. List those locations where visual inspection will be utilized and those locations where measurements will be taken and also the associated acceptance criteria. A commitment should be included that the NRC will be provided documentation of any corrective action resulting from the tests and confirmation by additional testing that substantiates effectiveness of the corrective action.

RESPONSE

To provide more information regarding the testing programs and modes of operation, FSAR Section 3.9(B).2, Pages 3.9(B)-5, and 3.9(B)-6, has been revised.

More specific information concerning the locations where visual inspection or measurements are to be taken are addressed in the applicable test procedures. Acceptable criteria for the thermal and dynamic tests are addressed in the applicable FSAR Chapter 14.0 test abstracts.

Corrective action for any deficiency identified as a result of the test program will be available for inspection at the site. Retesting will be performed in accordance with administrative controls identified in Chapter 14.0.

time dependent forcing function, such as fast valve closure, while the second is a constant vibration, usually flow induced.

a. Transient response

Dynamic events falling in this category are anticipated operational occurrences. The systems are operated in their normal mode (emergency mode for auxiliary feedwater turbine pump), and measurements are recorded on the systems during and following the event that causes the transient induced vibrations. The systems and the associated transients to be included in the preoperational test program to verify the piping system are:

1. Main steam

(a) Main steam turbine stop valve trip

(b) Main steam atmospheric dump valves opening

(c) Main steam condenser dump valves opening

~~2. Reactor Coolant System~~

(a) Pressurizer power-operated relief valve ~~blowdown~~ piping

2. (a) Relief Valve Operation

3. Auxiliary Turbine System

(a) Auxiliary feedwater pump turbine stop valve trip

Selected snubbers subjected to the above transients are monitored during this preoperational testing to assure proper snubber operation.

All of the above are upset transients and a time dependent dynamic analysis is performed on the system. The stresses thus obtained are combined with system stresses resulting from other operating conditions in accordance with the criteria provided in Table 3.9(B)-2.

b. Steady state vibration

System vibration resulting from flow disturbances falls into this category. Positive displacement pumps may cause such flow variation and vibrations and, as such, will be reviewed. Such systems will be checked, including the charging systems.

Since the exact nature of the flow disturbance is not known prior to pump operation, no analysis is performed. A visual steady state vibration inspection is made during system operation. Measurements above the following guidelines are recorded:

> Frequency ≥10 Hz
> For safety-related systems ≥0.125 inches
> For nonsafety-related systems ≥0.25 inches

except the reactor coolant system and pressurizer surge line

Safety-related systems and high energy systems will be monitored for steady state vibration for all modes of system operation encountered during the preoperational test program defined in FSAR Chapter 14.0. For specifics of this testing, see FSAR Chapter 14.0 of each site addenda.

, including associated instrumentation,

(which corresponds to 10^6 cycles)

The acceptance criterion is that the maximum measured amplitude shall not induce a stress in the piping system greater than one-half the endurance limit, as defined in Section III of the ASME Boiler and Pressure Vessel Code, 1974.

When required, additional restraints are provided to reduce the stresses to below the acceptance criterion levels.

During the thermal expansion test, pipe deflections will be recorded at selected locations. The system will also be visually monitored for hanger and snubber performance and for piping interferences with structure or other piping. One complete thermal cycle, i.e., cold position to hot position to cold position, will be monitored.

Selected portions of the following systems will be monitored during their normal mode of operation.

Main steam system
Main feedwater system
Letdown/charging system
Residual heat removal system
Containment spray system (1)
Emergency core cooling system
Auxiliary feedwater system
Auxiliary turbine system
Steam generator blowdown system

3.9(B).2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

3.9(B).2.2.1 Safety-Related Equipment in the NSSS

Refer to Section 3.9(N).2.2.

3.9(B).2.2.2 Safety-Related Mechanical Equipment Other Than the NSSS

The criteria used to decide whether dynamic testing or analysis should be used to qualify seismic Category I mechanical equipment are as follows:

a. Analysis without testing

(1) Design characteristics of the containment spray system do not permit actual testing to monitor thermal expansion of the suction piping from the containment sumps, during the recirculation mode. Verification of this piping will be attained by its similarity to the RHR suction lines from the RCS hot leg which will be monitored.

#20 Section 3.9(B).2.1, Page 3.9(B)-5

The applicant should indicate whether the listed systems meet the SRP 3.9.2 requirements with respect to the scope of this program.

RESPONSE

Our modified program satisfies SRP 3.9.2 requirements.

#21

Provide assurance that the functional capability of all ASME Class 1, 2 and 3 piping systems essential to plant safety is maintained under all designated loading conditions.

RESPONSE

For faulted condition analysis of Class 1 branch piping attached to the reactor coolant loop, Equation (9) of ASME Section III, Subsection NB-3652 is applied with a stress limit of 3.0 Sm. This criterion provides sufficient assurance that the piping will not collapse or experience gross distortion such the the function of the system would be impaired. The basis for this position is described in the Westinghouse response to NRC Question 110.34 on the RESAR-414 application (Docket No. STN 50-572), which received a Preliminary Design Approval (PDA) in November 1978.

For Class 2 and 3 piping systems 2-1/2" and larger, the MEB Regulatory Position in "Interim Technical Position - Functional Capability of Passive Piping Components" dated 07/19/78 is met. This has been verified through the use of the Bechtel computer code ME-101 which is described in Section 3.9(B).1.2.1.2 of the FSAR.

For small bore (i.e. 2" normal diameter and smaller) Class 2 and 3 piping, a standard Bechtel program us used to assure that the ASME code requirements are met. The results of the program have been shown to be conservative when compared to the results of ME-101. Since ME-101 assures the functional capability of lareg bore piping and the standard Bechtel program is conservative when compared to ME-101, the functional capability of small bore piping is assured.

Small bore piping is designed in the design office and shown on the SNUPPS model. Therefore, the analyzed design is the actual design installed in the field and the analyses properly consider the final design (i.e. routing, hanger locations, temperature, concentrated masses, etc).

All small bore piping has a D_O/t ratio less than 50, which ensures stability and no local buckling.

22 Section 3.9(B).3.3.1.g, Page 3.9(B)-15

Please list all instances when a dynamic load factor of less than 2.0 was used and provide the needed justification.

RESPONSE

For all systems analyzed by static methods, a dynamic load factor of two has been used. A dynamic load factor was not used for those systems which were analyzed dynamically.

d. Where more than one safety or relief valve is installed on the same run pipe, the sequence of valve openings which induce the maximum stresses is considered as required by Regulatory Guide 1.67.

e. The minimum moments to be used in stress calculations are those specified in ASME Code Case 1569.

f. The effects of the valve discharge on piping connected to the valve header are considered.

g. The reaction forces and moments used in stress calculations include the effects of a dynamic load factor (DLF) or are the maximum instantaneous values obtained from a dynamic time-history analysis. A dynamic load factor of 2.0, as required by Regulatory Guide 1.67, is used when a system is analyzed by static methods.

3.9(B).3.3.2 Closed Discharge

A closed discharge system is characterized by piping between the valve and a tank or some other terminal end. Under steady-state conditions, there are no net unbalanced forces. The initial transient response and resulting stresses are determined, using either a time-history computer solution or a conservative equivalent static solution. In calculating initial transient forces, pressure and momentum terms are included. If required, water slug effects are also included.

3.9(B).3.3.3 Operational Qualification for Active Safety-Relief Valves

Active safety-relief valves are subjected to the following shop tests, hydrostatic, seat leak tests, and a static loading equivalent to the SSE applied at the top of the bonnet and pressure at the valve inlet increased until the valve mechanism actuates. Periodic in situ valve inspection is performed to assure the functional ability of the valves.

During a seismic event, it is anticipated that the seismic accelerations imposed upon the valve may cause it to open momentarily and discharge under system conditions which otherwise would not result in valve opening. This is of no real safety or other consequence.

3.9(B).3.4 <u>Component Supports</u>

3.9(B).3.4.1 Supports Furnished with the NSSS

Refer to Section 3.9(N).3.4.

#23 Section 3.9(N).2.1, Page 3.9(N)-33

Please describe the acceptance limits that will be used for visual inspection of vibration. How will the stresses associated with the vibration be calculated? What ASME Code stress and fatigue limits will be used? What measures will be taken to monitor the thermal movement of the primary loop during heat up to ensure that no restraint to thermal growth is encountered?

RESPONSE

SEE REVISED page 3.9(N)-33.
Westinghouse will monitor the thermal motion of the RCL during hot functional testing and will provide shim sizes to assure proper clearances and support are provided for the RCS components.

c. Component support buckling allowable load

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 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. In no case shall the compressive load exceed 0.9 times the critical buckling strength.

Loading combinations and allowable stresses for ASME Code, Section III, Class 1 components and supports are given in Tables 3.9(N)-2 and 3.9(N)-3. For faulted condition evaluations, the effects of the SSE and LOCA are combined using the square-root-of-the-sum-of-the-squares method. Justification for this method of load combination is contained in References 4 and 5.

3.9(N).2 DYNAMIC TESTING AND ANALYSIS

3.9(N).2.1 Preoperational Vibration and Dynamic Effects Testing on Piping

A preoperational piping vibrational and dynamics effects testing program will be conducted for the reactor coolant loop/supports systems during startup functional testing of the SNUPPS units. The purpose of these tests will be to confirm that the systems have been adequately designed and supported for vibration as required by Section III of the ASME Code, paragraphs NB-3622.3. The tests will include reactor coolant pump starts and trips. If vibrations are observed which, from visual examination, appear to be excessive, either: 1) an instrumented test program will be conducted and the system reanalyzed to demonstrate that the observed levels do not exceed one half the endurance limit, at 10^6 cycles, defined in the ~~cause~~ ASME Code ~~stress and fatigue limits to be exceeded~~, 2) the cause of the vibration will be eliminated, or 3) the support system will be modified to reduce the vibrations. Particular attention will be provided at those locations where the vibrations are expected to be the largest for the particular transient being studied as per the criteria of the ASME Code as referenced above.

It should be noted that the layout, size, etc., of the reactor coolant loop and surge line piping used on SNUPPS is very similar to that employed in Westinghouse plants now in operation. The operating experience that has been obtained from these plants indicates that the reactor coolant loop and surge line piping are adequately designed and supported to minimize

#2(Section 3.9(N)-2.4, Page 3.9(N)-36

The FSAR should clearly state that the SNUPPS plants are classified as non-Prototype Category I in accordance with Reg. Guide 1.2(.

RESPONSE

A correlation between SNUPPS vibration predictions and prototype testing was discussed in detail. The prototype plant for SNUPPS is Indian Point Unit 2. This plant (Indian Point) was fully instrumented and tested during hot functional and initial startup testing. Data applicable to SNUPPS were also obtained from tests on the Trojan 1 and Sequoyah 1 plants. The significant differences between SNUPPS and Indian Point internals are the replacement of the annular thermal shield with neutron panels, modifications resulting from the use of 17 X 17 fuel, and the change to the UHI-style inverted top hat upper internals.

FSAR Section 3.9(N).2.3 will be revised to address the correlation between SNUPPS and prototype internals vibration testing.

#25 Table 3.9(N)-3

The appropriate ~~sections~~ paragraphs of ASME Section III should be referenced for the various components listed.

RESPONSE

The applicable subsections of the ASME Code which applies to Class 1 components are as follows:

Vessels / Pumps / ~~Valves~~	NB 3200
Valves	NB 3500
Piping	NB 3600
Component Supports	NF 3000

Table 3.9 (N)-3 will be revised to identify the specific Code paragraph and service level applicable to Class 1 components and piping.

26 Section 3.6.2.1.1.9.2(B), Page 3.6-10

The pipe break criteria is not in compliance with SRP 3.6.2 in that the 3.0 S_m value should be 2.4 S_m.

RESPONSE

See revised Section 3.6.2.1.1.a.2.

in order to verify the design basis break locations in the reactor coolant loop noted therein.

At all postulated circumferential break locations, the maximum loop piping displacements, as determined by the dynamic RCS analysis or the location of pipe restriants, are such that the separation results in a limited flow area. Longitudinal breaks are assumed to have an opening area equal to one flow area of the pipe.

2. Pipe breaks are postulated to occur in the following locations in Class 1 piping runs or branch runs outside the primary reactor coolant loops and pressurizer surge line as follows:

   (a) The terminal ends of the piping or branch run.

   (b) Any intermediate locations between the terminal ends where stresses, calculated using equations (12) and (13) of the ASME B&PV Code, Section III, Subsection NB, exceed 2.4 Sm, where Sm is the design stress intensity, as given in the ASME B&PV Code, and the stress range calculated, using equation (10) of the ASME B&PV code, exceeds 2.4 Sm.

   (c) Any intermediate locations between terminal ends where the cumulative usage factor, derived from the piping fatigue analysis, under the loadings associated with the OBE and operational plant conditions, exceeds 0.1.

   (d) Additional locations of maximum stress intensity or cumulative usage factor to assure a minimum of two break locations between terminal ends.

A complete discussion of the reactor coolant loop break locations is provided in Reference 1.

b. ASME B&PV code, Section III - Class 2 and 3 Piping Within Protective Structures

   1. Breaks are postulated to occur at terminal ends, including:

      (a) Piping-pressure vessel or equipment nozzle intersection

      (b) High-energy/moderate-energy boundary

      (c) Pipe to pipe anchor intersection

Rev. 2
2/81

#21 Section 3.6.2.1.1.e, Page 3.6-13

Please provide details of all locations where welded attachments were made to portions of piping covered under this section.

RESPONSE

Welded attachments have not been used on systems falling in this category, i.e. high-energy piping in containment penetration areas. Location details will be provided if welded attachments are used in the future.

#28 Clarify how loads are combined (e.g. absolute sum, SRSS, etc.).

RESPONSE

The methodology of load combinations and applicable stress limits were discussed. In particular, the following items were noted:

1.For primary equipment, primary equipment supports, and Class 1 branch lines, LOCA and SSE were combined by SRSS on a load component basis (the LOCA and SSE forces in the x direction were combined by SRSS, the LOCA and SSE moments in the x direction were combined by SRSS, etc.).

2.For RCL piping, the deadweight moments were added to the LOCA moments prior to the SRSS combination of the LOCA and SSE loads. An evaluation was performed to show that if the deadweight moments were added to the SRSS of the LOCA and SSE (per NUREG-0484), the maximum loop stresses would increase by less than 0.2%. It was noted that the deadweight moments are approximately two orders of magnitude less than either the LOCA or SSE moments.

3.For Class 2 and 3 equipment, the loads identified in Table 3.9(N)-4 are combined by absolute sum.

The FSAR will be clarified as necessary for the applicability of the load combination and stress limit tables and the FSAR will include the load combination methodology applicable to Class 1 and Class 2 & 3 components and supports. Additionally, the limits used for component support bolting will be added to the FSAR.

4. The faulted condition stress limits for ASME Code Class 1 component support bolting are defined in the Westinghouse position on Regulatory Guide 1.124 contained in RESAR 414. This position was previously accepted by the staff.

5. For primary equipment supports, the criteria of F-1370(d) ($2/3\,P_c$) are met.

#29 3.9.6

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems thus causing an inter-system LOCA.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specifications which will require corrective action; i.e., shutdown or system isolation when the final approved leakage limits are not met. Also, surveillance requirements, which will state the acceptable leak rate testing frequency, shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average to be approximately one year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance, etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute for each valve (GPM) to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting valve would be an indication of valve degradation from one test to another.

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM above the previous test leak rate or system design precludes measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case by case basis.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves. Also discuss in detail how your leak testing program will conform to the above staff position.

RESPONSE

FSAR Section 5.2.5.2.1 provides a discussion of those auxiliary systems that interface with the reactor coolant system. The reactor cooling system is shown on FSAR Figure 5.1-1.

The pump and valve inservice inspection and test program will be submitted in July 1981. This will include the valves which separate the RCS from low pressure systems. However, a separate submittal concerning isolation of the RCS from low pressure systems will also be made in July 1981.

30 GENERAL

The applicant has not yet submitted his program for preservice and inservice testing of pumps and valves.

RESPONSE

The pump and valve inservice inspection and test program will be submitted in July 1981.

#31 TO ALL APPLICANTS:

Due to a long history of problems dealing with inoperable and incorrectly installed snubbers, and due to the potential safety significance of failed snubbers in safety related systems and components, it is requested that maintenance records for snubbers be documented as follows:

Pre-service Examination

A pre-service examination should be made on all snubbers listed in tables 3.7-4a and 3.7-4b of Standard Technical Specifications 3/4.7.9. This examination should be made after snubber installation but not more than six months prior to initial system pre-operational testing, and should as a minimum verify the following:

(1) There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

(2) The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifications.

(3) Snubbers are not seized, frozen or jammed.

(4) Adequate swing clearance is provided to allow snubber movement.

(5) If applicable, fluid is to the recommended level and is not leaking from the snubber system.

(6) Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational test exceeds six months due to unexpected situations, re-examination of items 1, 4, and 5 shall be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements must be repaired or replaced and re-examined in accordance with the above criteria.

Pre-Operational Testing

During pre-operational testing, snubber thermal movements for systems whose operating temperature exceeds 250 F should be verified as follows:

(a) During initial system heatup and cooldown at specified temperature intervals for any system which attains operating temperature, verify the snubber expected thermal movement.

(b) For those systems which do not attain operating temperature, verify via observation and/or calculation that the snubber will accommodate the projected thermal movement.

(c) Verify the snubber using clearances at specified heatup and cooldown intervals. Any discrepencies or inconsistencies shall be evaluated for cause and corrected prior to proceeding to the next specified interval.

The above described operability program for snubbers should be included and documented by the pre-service inspection and pre-operational test programs.

The pre-service inspection must be a prerequisite for the pre-operational testing of snubber thermal motion. This test program should be specified in Chapter 14 of the FSAR.

RESPONSE

Pre-Service Examination

The concerns of items 1, 2, 4, and 6 will be satisfied under 79-14 walkdown procedure. Item 3 will be demonstrated prior to the 79-14 walkdown inspection. The steam generator upper supports are the only hydraulic snubbers used. Item 5 will be satisfied for these snubbers by a preservice inspection.

Pre-Operational Testing

During the thermal expansion test, snubber movements will be verified by recording the deflections in the pipes. Also, the system will be visually monitored for snubber performance and for piping interference with structure or other piping for one complete thermal cycle. The cause of any deficiency will be evaluated and corrected accordingly.

If the period between the initial pre-service examination and initial system pre-operational test exceeds six months, re-examination of items 1, 4, and 5 will be performed.

SNUPPS

#32 What is the SNUPPS position on Regulatory Guide 1.121 ?

RESPONSE

The SNUPPS position on Regulatory Guide 1.121 is contained in FSAR Appendix 3A. Reg Guide 1.121 analyses for the Model F steam generator have not been completed. Upon completion, tech spec limits will be evaluated and the information added to the Reg Guide 1.121 position.

SNUPPS

#33 Justify not analyzing for the effects of longitudinal pipe breaks in Class I branch lines.

RESPONSE

The effects of longitudinal breaks in Class I piping were not analyzed because of the seamless construction of the pipe.

SNUPPS agreed to provide additional justification for this exception to MEB 3-1. The axial stress to circumferential stress ratio will be considered.

#34 Provide W position on Reg. Guides 1.124 and 1.130.

RESPONSE

Appendix 3A will be revised to provide the position on these Guides for the NSSS. This position will be essentially the same as that approved by the NRC for RESAR 414. Additionally, the tables ~~in~~ 3.9(N)-1 will be revised to reference appendix 3A for Reg. Guides 1.124 and 1.130.

#~~36~~ 35 Reference to Section 5.4.1 in 3.9(N).3.3.A is not correct

RESPONSE

Section 3.9(N).3.3.A will be revised to delete the reference to 5.4.1 and state that the only source of vibration is the reactor coolant pumps.

#36 In Table 3.9(B) - 7, does the sum of stresses acting during a faulted event exceed 2.4 Sh?

Response

Table 3.9(B) - 7 will be modified for faulted conditions as follows:

The sum of stresses due to internal pressure, live and dead loads, and those due to occasional loads identified in the Design Specification as acting during a faulted event will not exceed 2.4 times the allowable stress S .

#37 Clarify FSAR Tables 3.9(B) -3 and -5 to distinguish which ASME paragraphs apply to divisions 1 and 2.

Response

Tables 3.9(B) -3 and -5 will be clarified as discussed in the meeting.

#38 On page 3.9(N) - 44, 2nd line, clarify the method by which loads were combined.

Response

Section 3.9(N).2.5 of the FSAR will be revised (page 3.9(N) - 44, 2nd line) to read . . . " combined (by SRSS) . . . "

#39 What allowable stresses were used for anchor bolts used in BOP?

Response

The following change will be made to Section 3.9.

All ASME Section III, Class 2 and 3, supports are designed as welded attachments to embedded or surface mounted plates. Bolting for the plates is designed according to AISC allowables with increases allowed by the loading case identified in FSAR Table 3.8-5. In no case do the tensile stresses in bolts exceed the yield stress of the bolting material at temperature.