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Docket Nos. 50-348
50-364

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Joseph M. Farley Nuclear Plant
Response to Request for Additional
Information Concerning NUREG-0737 Item II.D.1

Gentlemen:

By letter dated May 24, 1993 the NRC staff requested additional information regarding Southern Nuclear Operating Company's (SNC) August 7, 1992 response to NUREG-0737 Item II.D.1, "Performance Testing of BWR and PWR Relief and Safety Valves." In the attachment to this letter SNC addresses each of the Staff's questions. If any additional information is needed to complete your review of this matter, please advise.

Respectfully submitted,

D. Morey

Dave Morey

Attachment

BHW:sar INQRES.DOC

cc: Mr. S. D. Ebneter
Mr. T. A. Reed
Mr. M. J. Morgan

A046
1/1

Attachment

Inquiry No. 1

The August 7, 1992, submittal, identifies that the computer program ITCHVENT was used for the thermal hydraulic analysis. Since the submittal describes this program as an updated version of the program ITCHVALVE, which had been previously benchmarked against the EPRI test data, describe any changes incorporated in ITCHVENT which could impact the original benchmarking process.

Response No. 1

Program ITCHVALVE is a one-dimensional thermal hydraulic transient analysis code using the method of characteristics to carry out its numeral calculations. The types of fluids that the code can analyze are homogeneous, equilibrium flow. The fluids considered in the program are homogeneous initially. However, after discharge, the fluids are dispersed at the valves. Mixing of the water and steam is considered in the analysis. In addition, the code can treat characteristics when they cross an interface that is perpendicular to the flow. The code offers a choice of schemes for computing the material properties. One of the choices is a mixed implicit/explicit material characteristics computation. The other is a finite difference mass and energy conservation equation. These are complex and conservative methods used to solve the time history thermal hydraulic transient problems.

Additional efforts were put in to improve the program capabilities in early 1992 and the program was then renamed ITCHVENT. The upgraded version of the program ITCHVENT enhanced the capability to solve a variety of cases. It also improved the efficiency of the time-history integration. The updated version of the program ITCHVENT was also verified and benchmarked against the EPRI test data. The results of the verification and benchmarking process were found to be acceptable. Documentations were filed in Westinghouse Central File.

Program ITCHVENT/ITCHVALVE, which generates thermal hydraulic forcing functions, was benchmarked to the EPRI test results for test 908 (cold loop seal) and test 917 (hot loop seal). Attached in Figures 1 and 2 are comparisons of the EPRI test results with the thermal hydraulic analysis results performed by ITCHVALVE. The analysis results generated by ITCHVENT were also compared to the ITCHVALVE and EPRI results and were found to envelope the ITCHVALVE results. Figures 3 and 4 also demonstrate the shapes of the forcing functions are very similar.

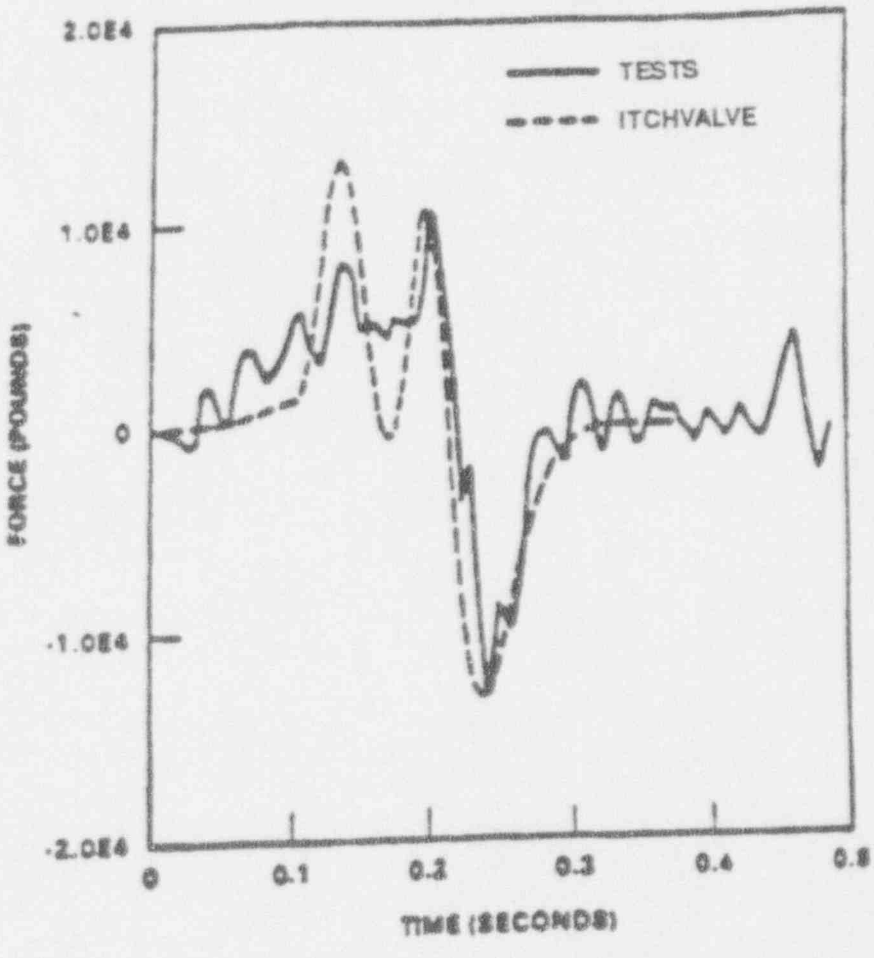

Figure 1

Comparison of the EPRI Force Time-History for WE32 and WE33 from Test 917 with the Thermal Hydraulic Analysis Predicted Force Time-History

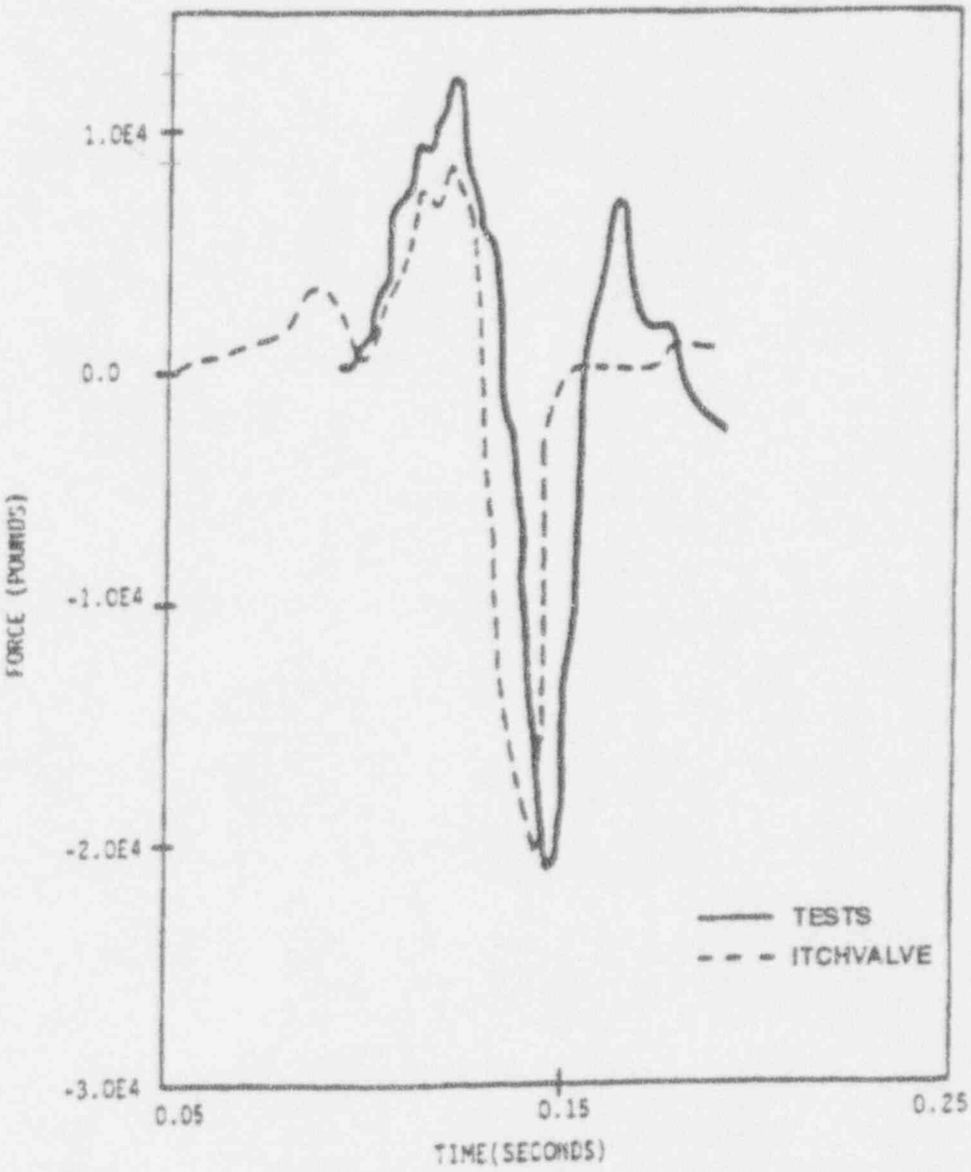

Figure 2

Comparison of the EPRI Force Time-History for WE28 and WE29 From Test 908 with the Thermal Hydraulic Analysis Predicted Force Time-History

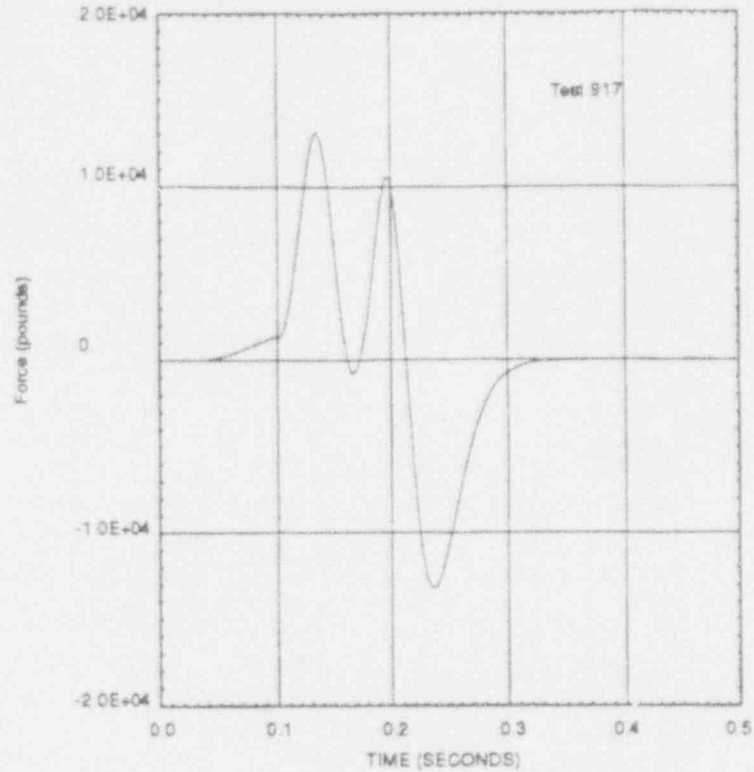

Figure 3. Results from Program ITCHVALVE

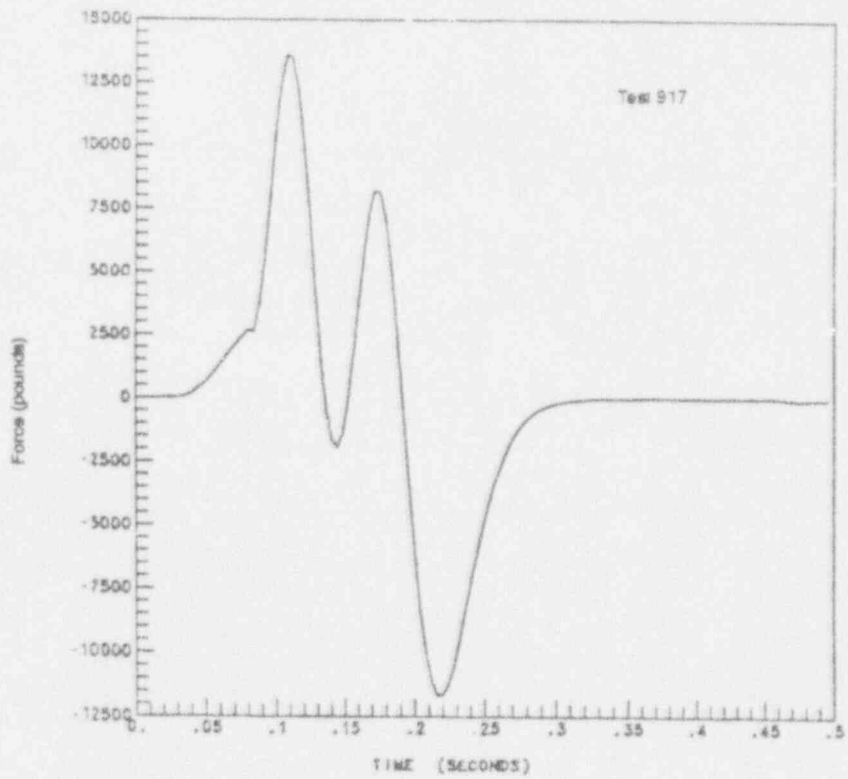

Figure 4. Results from Program ITCHVENT

Inquiry No. 2

The submittal also identifies that the computer program WECAN was used to perform an elastic/plastic analysis of the pressurizer safety valve piping system. Provide a description of the benchmark procedure for this computer program. This description should include the comparisons made between the results of this program and actual test data. The description of these comparisons should include the relevance of the benchmark data to the analysis of the safety valve discharge piping analysis.

Response No. 2

The program WECAN is a general purpose finite element structural analysis program developed at Westinghouse to perform static and dynamic linear and nonlinear structural analyses. This program was certified and benchmarked in WCAP-8929, April 1977 (Reference 1) in compliance with criterion number 1 in GDC Appendix A and the Design Control measures in "Quality Assurance for Nuclear Power Plants" of Appendix B in 10 CFR 50 of the NRC rules and regulations. The benchmark procedure can also be found in this WCAP. In addition to this WCAP report, WECAN was also benchmarked for the elastic-plastic type of elements used in the Farley piping analysis. Documentations were filed in the Westinghouse central file. Benchmarking of WECAN against the specific pressurizer safety valve discharge test data is not required since the thermal hydraulic forcing function code ITCHVALVE/ITCHVENT was benchmarked to the EPRI test data.

Reference:

1. Westinghouse WCAP-8929 "Benchmark Problem Solutions Employed for Verification of the WECAN Computer Program," April 1977.

Inquiry No. 3

Explain how the emergency load limits specified in Tables 3 and 4 of the August 7, 1992, submittal, have been met. If the elastic/plastic analysis was used to satisfy this load combination, provide a discussion of how the elastic/plastic analysis results meet the applicable Code criteria for the Emergency condition.

Response No. 3

(NOTE: For convenience, Tables 3, 4, and 6 of the August 7, 1992, submittal are included in this letter as Tables 1, 2, and 3, respectively.)

It should be emphasized that the pressurizer safety and relief valve piping (PSARV) was originally qualified to its design basis allowables prior to the TMI incident. The design basis as the ASME Code Section III, 1971 edition, including summer 1971 addenda for the Ci ss 1 portion of the system and the ANS B31.1 - 1967 Code with the 1971 addenda for the NNS portion of the piping. Only after the TMI incident and the subsequent requirements of Reference 1 were the more conservative limits listed in Tables 1 and 2 required, respectively, for Class 1 and NNS piping of the PSARV system. More importantly, these conservative limits are not ASME Code requirements but they are based on the recommendations from a piping subcommittee of the PWR PSARV test program.

From the overall safety viewpoint, the elastic analysis includes many conservative assumptions including the simultaneous opening of all three safety valves. The analysis demonstrates that (i) all three safety values remain operable, (ii) all Class 1 portion of the piping system satisfy the limits in Table 1, and (iii) the structural integrity of the NNS portion of the piping system will be maintained. Therefore, it is our belief that while the elastically calculated stress in the NNS portion exceed conservative allowables of Table 2, the structural integrity of the NNS piping can be demonstrated by maintaining the pressure boundary, delivering the flow, and maintaining its functionality through elastic/plastic analysis.

Table 3 has listed the elastically calculated stresses in both the Class 1 and NNS portions of the piping system along with the respective allowable stresses for emergency conditions. All points in Class 1 met the emergency condition allowables. For some high stress points in NNS piping, the elastically calculated stresses in a branch connection (6" x 12" x 12" - with conservative stress intensification factor of 2.2) and at a reducer (6" x 12" - with conservative stress intensification factor of 2.0) exceeded the emergency condition allowable stresses.

To demonstrate the structural integrity in the NNS piping of the system, the complete safety valve discharge piping system was modeled using WECAN elastic/plastic straight and curved pipe elements and loaded with hydraulic force time-histories from thermal hydraulic analysis results (output of ITCHVENT). The resulting maximum total strains from this elastic/plastic analysis at the fore-mentioned branch connection and reducer are 0.121% and 0.228%, respectively. These are small values for total strain (elastic plus plastic) in comparison with the allowables listed by Code Case N-47 for emergency conditions (2% for membrane plus bending and 5% for local). Even more importantly, the calculated maximum total strain presented above was at a single point from an eight points circumference of a cross-section in the WECAN super elements. All other seven points have much smaller values with or without plastic strains such that no cross-sectional area change was possible. Based on these small values in total strain, it is our judgment, with or without using Code Case N-47 guidelines, that the NNS piping system can maintain its structural integrity, deliver the flow and maintain functionality.

Reference:

1. NUREG-0737, "Clarification of TMI Action Plant Requirements," NRC, Nov. 1980.

TABLE 1

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR PRESSURIZER SAFETY AND RELIEF VALVE PIPING - UPSTREAM OF VALVES CLASS 1 PIPING

Plant/ System Operating Condition	Load Combination	Piping Allowable Stress Integrity
Normal	N	1.5 S_m
Upset	N + OBE	1.5 S_m
Upset	N + SOT_U	1.5 S_m
Upset	N + OBE + SOT_U	1.8 S_m/1.5 S_y
Emergency	N + SOT_E	2.25 S_m/1.8 S_y
Faulted	N + SSE + SOT_F	3.0 S_m

TABLE 2

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA FOR PRESSURIZER SAFETY AND RELIEF VALVE PIPING - DOWNSTREAM OF VALVES
NNS PIPING

Plant/ System Operating Condition	Load Combination	Piping Allowable Stress Integrity
Normal	N	1.0 S_h
Upset	N + OBE	1.2 S_h
Upset	N + SOT_U	1.2 S_h
Upset	N + OBE + SOT_U	1.8 S_h
Emergency	N + SOT_E	1.8 S_h
Faulted	N + SSE + SOT_F	2.4 S_h

TABLE 3

FARLEY UNITS 1 AND 2
SAFETY LINE PIPE STRESS AND STRAIN SUMMARY
FOR EMERGENCY CONDITION

Node Point	Piping Components	Code Maximum Stress (ksi) •	Code Allowable Stress (ksi)	Total Strain (%) +
1290*	Butt Weld	18.15	36.225	~0
1460*	Long-radius elbow	31.65	36.225	0.15%
100**	Branch Connection	49.25	33.84	0.26%
690**	Reducer	71.08	33.84	0.46%
1490**	Welded Attachment @ support R120	54.97	55.42 ♦	---

* ASME Class 1 piping, upstream of safety valves
** ASME NNS piping, downstream of safety valves
• From elastic analysis
\+ From elastic -plastic analysis
♦ Based on ASME Code Case N-318 allowable

Inquiry No. 4

The submittal references ASME Code Case N-47 as the basis for the allowable strains. This Code Case was developed for elevated temperature design. In addition, this Code case has not been endorsed by the staff in Regulatory Guide 1.84 which contains a list of Code Cases the staff finds acceptable. Explain how the criteria used in the elastic/plastic analysis meets the applicable Code criteria that has been endorsed by the regulations or other staff guidance.

Response No. 4

An important point which must be noted is that only a few points in the NNS piping do not meet the emergency allowable stresses which represent additional conservative requirements since the plant original design basis was developed under ANS B31.1. It is this NNS portion of the system that Code Case N-47 was used, rather than the ASME Class 1 portion of the system.

Current practice for the design of nuclear piping based on the ASME Boiler and Pressure Vessel Code approach does not correlate well with the observed behavior of piping systems under dynamic loadings. The design limits are stress controlled and do not consider the plastic behavior of piping systems. Recent studies such as NUREG 1061, Volume 2 (Ref. 1), tests such as ANCO's piping system tests (Ref's. 2 and 3) and Imazu, Teidoguchi, Greenstreet elbow tests (Ref. 4, 5, and 6) indicate that the failure mode for piping is strain-controlled and the loading to failure defined as excessive flow area reduction is many times greater than the Code allowable values.

Code Case N-47 (Ref. 7) addresses the design and analysis of Class 1 components at elevated temperatures which exceed those listed in Section III, Appendix I. In addition to the low temperature Section III failure modes, the rules of the Code Case guards against deformation-related failures, such as creep rupture from long-term loadings, creep fatigue failure, and loss of function due to deformation or gross distortion caused by incremental collapse and ratcheting. Appendix T of the Code Case provides the following strain limits for normal, upset, and emergency conditions: 1% membrane, 2% membrane plus bending, and 5% for local strain at any point with discontinuity.

The strain criteria in the Code Case was developed for use at elevated temperature where stress criteria cannot be applied. Technically, the strain criteria in the Code Case can also be applied to the dynamic analysis of piping system/components at low temperature. The bases for low temperature applications are discussed as follows:

- The strain limits established in the Code Case consider all strains, plastic strain as well as creep and ratcheting strain. They are intended to define acceptable performance of structures and components in terms of total strain. For elevated temperature applications the creep cycling and ratcheting are added concerns. Thus, the imposed limitation in defining the strain limits for elevated temperature applications should be viewed as an added conservatism for low temperature applications.

- The strain limits in the Code Case are set low enough that they effectively ensure that small deformation theory is applicable. The basis for low temperature rule is also small deformation theory.

- The strain limits in the Code Case are specified for normal, upset and emergency conditions. The use of these strain limits must allow for operation under these strains with no significant impact on the integrity of structures and components.

References:

1. NUREG 1061, Vol. 2, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Seismic Designs - A Review of Seismic Design Requirements for Nuclear Power Piping," Apr. 1985.

2. NUREG/CR-3893, "Laboratory Studies: Dynamic Response of Prototypical Piping Systems," prepared by ANCO Engineers, Inc. for the U.S. NRC and EPRI, Aug. 1984.

3. EPRI Report No. NP-3746, "Dynamic Response of Pressurized Z-Bend Piping Systems Tested Beyond Elastic Limits and with Support Failures," prepared by ANCO Engineers, Inc. for EPRI, Dec. 1984.

4. Imazu, Sahahibara, Nagota and Hashimoto, "Plastic Instability Test of Elbows Under In-Plane and Out-of-Plane Bending," Paper E6/5, Sixth SMIRT Conference, Paris, France, Aug. 1981.

5. Teidoguchi, H., "Experimental Study on Limit Design for Nuclear Power Facilities During Earthquake", Japanese Report 50-1705 issues to U.S. NRC, Feb. 1975.

6. Greenstreet, W. L., "Experimental Study of Plastic Response of Pipe Elbows", ORNL/NUREG 24, Feb. 1978.

7. ASME Boiler and Pressure Vessel Code, Code Case N-47-23, "Class 1 Components in Elevated Temperature Service, Section III, Division I", approved February 20, 1986.

Inquiry No. 5

The submittal states that the valve thrust loads and the SSE loads were combined by the square-root-of-the-sum-of-the-squares (SRSS) method. The staff's position regarding acceptable methodology for combining dynamic responses is contained in NUREG-0484. The SRSS method is limited to cases where the responses are calculated on a linear elastic basis because there is no technical basis for combining responses based on an elastic analysis with the results of an elastic/plastic analysis. Provide a technical justification for the load combination used.

Response No. 5

The SRSS method for combining dynamic loads contained in NUREG-0484 is limited to cases where responses are calculated on a linear elastic basis. As mentioned in NUREG-0484, from a study for a worst case model whose ductility ratio equals two, the dynamic reserve margin is 50% greater than the static reserve margin and the allowable stress limits are based on static load considerations. Hence the SRSS method for combining dynamic loads is very conservative. Studies also have been made to justify the decoupling of the Safe Shutdown Earthquake (SSE) and other dynamic loads. Depending on the operations procedures, the valve thrust discharge event may have no chance of occurring during the SSE earthquake or the probability is extremely low. Furthermore, the peak responses of these events may not happen at the same instance. Therefore, the SRSS method of combining these loads gives rise to very conservative loading input for the structural evaluation of the system.

In the Farley evaluation, the SRSS combination method was used to combine the elastically calculated SSE support loads to the elastic-plastically calculated valve thrust load at supports based on the following reasons:

- The magnitude of the SSE loads is small in comparison with valve thrust loads. The difference between SRSS or absolute sum combination is very small.

- The probability of a SSE event is approximate 10^{-4}/year. The probability of pressurizer safety valve operation is also extremely low since such operation requires the following infrequent events to occur simultaneously: loss of external loads and/or turbine trip, loss of normal feed water, and loss of all AC power to the station auxiliaries. Therefore, the probability of all of these events occurring at the same time is extremely small. Moreover, the probability of the largest peak of the SSE occurring at the same time as the largest peak of valve thrust with the same phase is much smaller than the two events occurring at the same time.

- From a practical engineering viewpoint, it is prohibitive to perform such a time history analysis by inputting both SSE time history and valve thrust time-history to the structure model and still maintain a conservative phasing relationship.

- Due to the small amount of the plasticity exhibited in the system under the valve thrust loading, the difference between support loads from elastic analysis and elastic-plastic analysis are small. Therefore, the SRSS combination is accepted in a similar manner as the combination of two elastic analysis results.

Inquiry No. 6

The submittal states that stresses in a few standard components exceed ASME Subsection NF and Appendix F criteria. Provide a list of these components which exceed the Code stress limits. Also provide the calculated stresses for these components.

Response No. 6

The terminology in the submittal regarding the stress level in the standard components may be misleading. There are five supports for which the manufacturer's standard Load Capacity Data Sheet (LCD) Level D allowables were exceeded. The "failed" supports are:

Unit 1 Supports	Unit 2 Supports
Tag RC-R54	Tag 2-RC-R104
Tag RC-R61	Tag 2-RC-R107
	Tag 2-RC-R133

The manufacturer for these components is Grinnell Corporation. Grinnell's Pipe Support Division was consulted and it was determined that in general LCDs, including the ones used for this effort, typically include conservatism with respect to ASME, Section-III, Subsection-NF and Appendix-F allowable stress limits.

Therefore, Grinnell was contracted to perform a detailed evaluation of each individual structural element of the subject components to determine the actual stress usage factor utilizing the ASME Section-III, Subsection-NF and Appendix-F allowable stress limits.

The results of the individual evaluations for the components are listed in Table 4 below:

TABLE 4

Support Tag	Component	Grinnell Part No.	Actual Load [kips]	Allowable Load [kips]	Usage Factor [Act/All]
	Hydraulic Snubber	Fig. 200 1-1 1/2" Dia	7.01	7.42	0.91
RC-R54	Rear Bracket 0°	Fig. 200	7.01	7.56	0.93
	Sway Strut	Fig. 211 Size 1	16.45	18.1	0.91
	Pipe Clamp 12" Dia.	Fig. 211 Size 1	16.45	23.7	0.69
RC-R61	Rear Bracket 45°	Fig. 211	16.45	17.4	0.95
	Sway Strut	Fig. 211 Size 2	20.5	23.8	0.86
2-RC-R104	Rear Bracket 0°	Fig. 211	20.5	29.3	0.70
	Sway Strut	Fig. 211 Size 3	27.0	36.3	0.74
	Pipe Clamp 12" Dia.	Fig. 211 Size 3	27.0	36.7	0.74
2-RC-R107	Rear Bracket 90°	Fig. 211	27.0	27.5	0.98
2-RC-R133	Pipe Clamp 6" Dia.	Fig. 211 Size 1	18.0	21.9	0.82

<u>Conclusion:</u>

In summary, the components failed with respect to standard manufacturer's LCD's, but were actually within allowable stress limits establish by the ASME Section-III, Subsection-NF, and Appendix-F code.

Inquiry No. 7

The submittal states that a factor of safety of 1.3 was used in lieu of 4.0 for concrete expansion anchor bolts. This is not acceptable to the staff. Those anchor bolts that do not meet the guidelines specified in IE Bulletin 79-02 should be modified to meet the bulletin requirements.

Response No. 7

Farley Units 1 and 2 PSARV discharge piping system consists of an ASME Class 1 portion and a Non-Nuclear Safety (NNS) portion. The Class 1 portion is supported by 6 Class 1 pipe supports in both units and the NNS portion is supported by 48 (Unit 1) and 52 (Unit 2) pipe supports.

Class 1 Supports - All ASME class 1 PSARV pipe supports, in Unit 1 and Unit 2, were evaluated and successfully qualified for the new loading combination in accordance with IE Bulletin 79-02.

There are a total of 48 Unit 1 NNS pipe supports, 21 of which have base plates with concrete expansion anchors. Unit 2 has 52 total NNS pipe supports, 25 of which have base plates with concrete expansion anchors. All base plates were evaluated in accordance with IE Bulletin 79-02. The results show that a small number of base plates have concrete expansion anchors (CEA) with factors of safety less than 4.0 when evaluated using the new loading condition (SSE combined with thrust loads). Tables 5 and 6 provide a detailed summary of the base plates and CEA factors of safety for Units 1 and 2, respectively.

The factors of safety for the Unit 1 and Unit 2 NNS pipe support base plates CEA should be considered acceptable for the following reasons.

- Original Design Basis 79-02 Program - FNP implemented a complete 100 percent verification testing effort in accordance with IE Bulletins 79-14 and 79-02 with strict quality control coverage. The completion of this effort minimizes the installation uncertainties and confirms the analytical methods which are the basis for the establishment of the minimum factor of safety of 4.0. Installation aspects such as verification of bolt torque, embedment depth, bolt spacing, minimum edge distance, nut engagement, and the as-built plate configuration, as well as accounting for plate flexibility in the analysis, have all been confirmed to be part of the base plate design basis analysis and qualification. All FNP base plates have been successfully analyzed and qualified for the original design loadings, (i.e., no overstresses were identified).

- NNS Class - The base plates which have factors of safety less than 4.0 are for the NNS class portion of the PSARV piping system.

- Conservative Load Combination - The load combination used in this evaluation includes the conservative combination of dead weight, maximum thermal expansion, and the SRSS combination of the SSE and the subject pressurizer safety valve (PSV) thrust loadings.

- Low Probability of Simultaneous Occurrence - The support loading combination includes the SRSS combination of SSE and PSV thrust loads. The thrust loads used are due to the simultaneous opening of the three pressurizer safety relief valves, and the subsequent discharge of three pipe loop seals. As outlined in the FNP Final Safety Analysis Report (Ref. 1) and Westinghouse WCAP-7769 (Ref. 2), there are three design basis events that could require the operation of the PSV: loss of external load and/or turbine trip, loss of normal feedwater, and loss of all AC power to the station auxiliaries. All three of these are classified as events of moderate frequency (i.e., expected frequency of occurrence is between once per year and once per 20 years). Furthermore, in order for PSV operation to be necessary during these events, there must be concurrent failures of the steam dump system, the pressurizer power operated relief valve, and the pressurizer spray system. Proper operation of any of these systems/components will preclude the actuation of the PSV.

  Considering the combination of the three low probability events occurring simultaneously with the concurrent failures of the secondary systems, PSV operation should be an extremely infrequent occurrence.

  References:

1. FNP FSAR, sections 15.2.7, 15.2.8, and 15.2.9.

2. Westinghouse WCAP-7769, Revision 1, "Overpressure Protection for Westinghouse Pressurized Water Reactors", Chapter 3.

TABLE 5
FARLEY NUCLEAR PLANT
UNIT 1 PSARV LINE PIPE SUPPORTS
ANCHOR BOLT DATA

Serial No.	Support Mark No.	Total No. of Bolts	No. of Bolts with F.S. ≥4	No. of Bolts with F.S. <4	Actual F.S. Bolt #	Actual F.S. F.S.	Types of Bolts with F.S. <4	Remarks
1	RC-R54	4	0	4	#1 #2 #3 #4	2.06 2.06 2.06 2.06	#1 THRU 4 3/4" DIA. HILTI KWIK	
2	RC-R56	8	1	7	#2 #3 #4 #5 #6 #7 #8	1.50 1.44 1.45 2.31 2.42 2.45 2.46	#2 THRU 8 7/8"DIA. HILTI KWIK	Two Plates 4 Bolts on Each Plate
3	RC-R61	4	2	2	#3 #4 #3 #4	1.40 2.65 OR 2.64 1.41	#3 AND 4 3/4"DIA. HILTI KWIK	Two Sets of Values of F.S Are Given, As All Forces and Moments are ±
4	RC-R70	10	8	2	#2 #5	3.73 2.99	#2-7/8" DIA. WEJ-IT #5-3/4"DIA. WEJ-IT	Two Plates One Plate with 6 Bolts One Plate with 4 Bolts #2 and 5 are on 6-Bolt Plate
5	RC-R99	4	3	1	#3	2.28	#3-3/4"DIA. HILTI KWIK	
6	RC-R215	8	6	2	#3 #7	3.92 3.07	#3 and 7 1" Dia. HILTI KWIK	Two Plates One Bolt on Each Plate

TABLE 6
FARLEY NUCLEAR PLANT
UNIT 2 PSARV LINE PIPE SUPPORTS
ANCHOR BOLT DATA

Serial No.	Support Mark No.	Total No. of Bolts	No. of Bolts with F.S. ≥4	No. of Bolts with F.S. <4	Actual F.S. Bolt #	 F.S.	Types of Bolts with F.S. <4	Remarks
1	2RC-R101	4	0	4	#1 #2 #3 #4	3.74 3.74 3.74 3.74	#1 THRU 4 1"DIA. HILTI KWIK	
2	2RC-R104	10	8	2	#3A #4A	3.85 3.89	#3A AND 4A 1 1/2"DIA.WEJ-IT	Two Plates 2 Bolts on the Same Plate
3	2RC-R110	8	4	4	#1 #2 #3 #4	2.58 2.58 2.58 2.58	#1 THRU 4 1"DIA. HILTI KWIK	Two Plates 4 Bolts on Same Plate
4	2RC-R111	4	0	4	#1 #2 #3 #4	3.15 3.15 3.15 3.15	#1 THRU 4 3/4"DIA. HILTI KWIK	
5	2RC-R112	4	0	4	#1 #2 #3 #4	2.63 2.63 2.63 2.63	#1 THRU 4 1/2"DIA. HILTI KWIK	
6	2RC-R115	4	2	2	#1 #3	3.54 3.54	#1 and 3 5/8"DIA. HILTI KWIK	
7	2RC-R117	2	0	2	#1A #2A	3.17 3.17	#1A and 2A 3/4"DIA. WEJ-IT	One Side Welded to Embed Plates
8	2RC-R120	6	2	4	#3 #4 #5 #6	1.94 1.94 1.94 1.94	#3 THRU 6 3/4" DIA. WEJ-IT	Two Plates Bolts 3 Thru 6 on Same Plate. One Side of Other Plate is Welded to Embed Plate

TABLE 6 (CONTINUED)
FARLEY NUCLEAR PLANT
UNIT 2 PSARV LINE PIPE SUPPORTS
ANCHOR BOLT DATA

Serial No.	Support Mark No.	Total No. of Bolts	No. of Bolts with F.S. ≥4	No. of Bolts with F.S. <4	Actual F.S. Bolt #	F.S.	Types of Bolts with F.S. <4	Remarks
9	2RC-R122X	2	0	2	#1 #2	2.72 2.72	#1 and 2 3/4"DIA. HILTI KWIK	One Side Welded To Embed Plate
10	2RC-R129X	10	4	6	#1 #2 #3 #4 #5 #6	3.62 3.58 3.66 3.56 3.39 3.87	#1 THRU 6 3/4"DIA. HILTI KWIK	Two Plates Bolts 1 Thru 6 are on Same Plate One Side of Other Plate Welded to Embed Plate
11	2RC-R131X	5	1	4	#2 #3 #4 #5	3.47 3.79 3.86 2.94	#2 THRU 5 1/2"DIA.HILTI KWIK	Welded to Embed Plate
12	2RC-R133X	8	0	8	#1 #2 #3 #4 #5 #6 #7 #8	2.76 2.76 2.76 2.76 3.43 3.43 3.43 3.43	#1 THRU 8 3/4"DIA. HILTI KWIK	Two Plates 4 Bolts on Each Plate
13	2RC-R139	2	1	1	#1	3.64	#1-1"DIA. HILTI KWIK	One Side Welded to Embed Plate