POWER AUTHORITY OF THE STATE OF NEW YORK

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(212) 397-6200

June 29, 1979
JPN-79-38

Director of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Thomas A. Ippolito, Chief
Operating Reactors Branch No. 3
Division of Operating Reactors

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Supplemental Information Supporting
Request for Plant Start-Up

Dear Sir:

A request for plant start-up about July 1, 1979 was submitted on June 8, 1979. This request was contingent upon completion of the piping stress reevaluation effort and any necessary modifications to constraints in inaccessible areas. A program plan was provided.

This supplemental information is submitted both as a result of discussions with the NRC staff (Messrs. Polk, Noonan, and Fair) on June 19, 1979 and to report progress made since June 8, 1979.

The Authority reaffirms its intent to complete the reevaluation effort of pipe stresses and pipe constraints in inaccessible areas and repeats its request for authorization to start up JAFNPP. The expected date is now July 15, 1979.

The status of the effort is shown below. Tabulations of the results are contained in Attachments 1 through 4.

1. PIPE STRESSES 271 076

All pipe stresses for the 96 piping lines are within allowable limits. Since the Authority's June 8 letter was submitted, branch piping having a moment of inertia greater

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than 10 percent of the run pipe moment of inertia were remodelled and incorporated into the reevaluation of the run pipe. As a result, 17 problems were recombined into 6 analysis packages and reevaluated for stresses. These packages are indicated in Attachment 1. This reevaluation again confirms the determination that the safety-related piping is within allowable limits. The tabulated pipe stress results are shown in Attachment 2.

2. PENETRATIONS

All loads on the 56 penetrations involved in the piping lines are within allowable limits (see Attachment 3).

3. NOZZLES

All loads on 50 of the 83 equipment nozzles involved in the piping lines are within allowable limits. The loads on the 33 remaining nozzles, subject to vendor confirmation, are also within allowable limits. Confirmation by the vendors is expected by July 2, 1979 (see Attachment 3).

4. PIPING SUPPORTS

All loads on 239 of the 342 piping supports in inaccessible areas are within allowable limits. Twenty-six supports tentatively are scheduled for modification and 77 supports remain to be evaluated.

Of 656 piping supports in the accessible areas, the loads on 243 are within allowable limits, and 12 are designated to be modified. Evaluation of the remaining 401 supports is continuing. The Authority's priorities for reevaluation in the accessible areas are noted in Attachment 1.

Of the pipe supports being modified, 9 pipe supports in the inaccessible areas and 5 in the accessible areas were designated for modification, not as a result of the pipe stress reevaluation, but as required to correct as-built deviations and for plant conformance with its as-designed basis.

In summary, of the total 998 piping supports involved, 482 have been evaluated acceptable and 24 are to be modified due to pipe stress reevaluation.

In order to estimate the number of structural modifications that might be required for pipe supports in the accessible areas, and to provide an assessment of the expected integrity of the accessible systems, a closer look was taken at the pipe supports which the Authority decided to modify due to pipe stress reevaluation in the inaccessible areas.

This examination tentatively indicates that:

1. As indicated in the June 8 letter, the stress analysis of piping incorporates numerous conservatisms for which no cumulative credit has been taken. (See Attachment 5)

2. Six of the supports are trunnions, which were originally designed utilizing basic analytical techniques and for which modifications are now being planned to satisfy local stress conditions. These trunnions are still acceptable based on the macro-stress criteria to which they were designed and will withstand the reevaluated loads.

3. One of the supports is within allowable loads for the combination of pressure + thermal + deadload + DBE conditions, although they have been designated for modification because they do not pass the OBE criteria. For this support, the OBE criteria was exceeded by 40 percent of the allowable.

4. Three supports are within 2.4 S_h (present ASME Code criteria for DBE). Five supports are not subjected to loads in excess of their ultimate strength and will retain their structural integrity under all projected load conditions.

   Of the remaining 2 supports being modified in the inaccessible areas, preliminary indications are that even if none of these supports provide any seismic restraint, the associated piping integrity will not be compromised.

Nothing in this analysis has identified any condition, as designed or built, which would endanger plant integrity or provide a basis for believing that JAFNPP would not withstand a seismic event. Final confirmation of the above will be transmitted prior to start-up.

Thus, extrapolation of the completed constraint evaluations to the accessible areas does not provide a basis for anticipating that any of the unresolved constraints in the accessible areas will render a critical system inoperable. On this basis, as well as that presented in the June 8 letter, the Authority believes

271 078

that JAFNPP should be permitted to start up prior to the completion of the seismic reevaluation flowing from the Show Cause Order.

After restart, if a constraint design evaluation indicates that such constraint may not be able to perform its intended function, and the Authority's analysis confirms that this inadequacy renders a safety-related system inoperable, the Authority will notify the NRC thereof within 24 hours of the determination. These safety-related systems are those which have plant shutdown requirements in the Technical Specifications in the event the system is declared inoperable. Repair of the deficiency shall be completed or justification for continued operation will be provided to the NRC, within 7 days or in accord with the appropriate plant technical specifications, whichever is less. If the above requirement cannot be met, the reactor will be placed in a cold shutdown condition within an additional 36 hours.

The completed reevaluation analyses and modification engineering documentation can be made available for NRC staff review. As the remaining analyses and modifications are completed, these will also be made available.

Finally, as requested by the NRC at the June 19 meeting, the effect of the reevaluation on the high energy pipe break analysis was investigated. As a result of this investigation, it was concluded that the original high energy pipe break analysis remains valid. A discussion of the high energy pipe break analysis is presented in Attachment 6.

Very truly yours,

Paul J. Early
Assistant Chief Engineer-
Projects

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IDENTIFICATION OF SYSTEMS AFFECTED AND PRIORITIES

The reevaluation effort included those piping systems originally computer analyzed with the SHOCK2 code and which perform or affect safety functions. The 96 original problems are identified in Appendix A were reevaluated using SHOCK3. The SHOCK3 problems are listed by piping system below, with the corresponding FSAR figure (flow diagram) noted. Seventeen of the SHOCK3 problems were subsequently recombined into 6 "branch problems" and reevaluated. These problems are identified in the table below as B1, B2, etc. Portions of non-safety piping systems were included in this effort where it was determined that such lines would affect the analysis of a safety related system. Examples include nonsafety interconnecting lines past the first automatic trip valve or the first normally closed manual valve and significant branch connections.

For the purpose of expediting startup of JAFNPP, the original 96 computer problems were divided into two priority classes for reevaluation. Priority 1 problems were those located in areas inaccessible during plant operation; these are areas where inspection and maintenance cannot be performed due to a high radiation environment. Priority 2 problems are those in

accessible areas; these systems were subdivided into priority subclasses A, B and C based on safety considerations. System accessibility and Priority 2 classification are also noted below.

System	Problem No.	Located in Inaccessible Areas	FSAR Fig. No.
Standby Gas Treatment	942	No (C)	5.3-2
	941	No (C)	5.3-2
Control Rod Drive	909	Yes	3.5-5
Residual Heat Removal	637	No (A)	4.8-1
	641 (B4)	No (A)	4.8-1
	643	No (A)	4.8-1

System	Problem No.	Located in Inaccessible Areas	FSAR Fig. No.	
	646 (B6)	No (A)	4.8-1	1.68
	647	No (A)	4.8-1	1.71
	650 (B3,B6)	Yes	4.8-1	1.75
	657	Yes	4.8-1	1.78
	664 (B2)	Yes	4.8-1	1.80
	682 (B2)	Yes	4.8-1	1.82
	737 (B3,B6)	Yes	4.8-1	1.84
	738 (B2)	Yes	4.8-1	1.86
	739	Yes	4.8-1	1.88
	757 (B3,B6)	Yes	4.8-1	1.90

271 082

System	Problem No.	Located in Inaccessible Areas	FSAR Fig. No.	
	948	No (A)	4.8-1	1.92
	951	No (A)	4.8-1	1.94
	888	No (A)	4.8-1	1.96
	867	No (A)	4.8-1	1.98
	870	No (A)	4.8-1	2.1
	871 (B6)	No (A)	4.8-1	2.3
	877	No (A)	4.8-1	2.5
	879	No (A)	4.8-1	2.7
	880 (B4)	No (A)	4.8-1	2.9
	864	No (A)	4.8-1	2.11

271 083

System	Problem No.	Located in Inaccessible Areas	FSAR Fig. No.	
	866	No (A)	4.8-1	2.13
	868	No (A)	4.8-1	2.15
	869	No (A)	4.8-1	2.17
	878 (B6)	No (A)	4.8-1	2.19
Standby Liquid Control	931	No (C)	3.9-1	2.22 2.23 2.24
Reactor Water Clean up	666	Yes	4.9-1	2.27 2.28 2.29
Reactor Core Isolation Cooling	656	Yes	4.7-1	2.31 2.32 2.33

271 084

System	Problem No.	Located in Inaccessible Areas	FSAR Fig. No.	
	667	Yes	4.7-1	2.35
	742	No (B)	4.7-1	2.37
	933	No (B)	4.7-1	2.39
Core Spray	651	Yes	7.4-6	2.41
	669 (B2,B3)	No (B)	7.4-6	2.43
	673	No (B)	7.4-6	2.45
	674	No (B)	7.4-6	2.47
	934	No (B)	7.4-6	2.49
Reactor Building Cooling Water	873	No (B)	9.7-1	2.51 2.52 2.53 2.54

POOR ORIGINAL

271 085

System	Problem No.	Located in Inaccessible Areas	FSAR Fig. No.	
	872	No (B)	9.7-1	2.56
Fuel Pool Cooling & Cleanup	949	No (C)	9.4-1	2.58 2.59
	950	No (C)	9.4-1	2.60
	952	No (C)	9.4-1	2.62
	953	No (C)	9.4-1	2.64
	947	No (C)	9.4-1	2.66
High Presure Coolant Injection	655	Yes	7.4-2	2.69 2.70 2.71
	679 (B1)	Yes	7.4-2	2.72
	681	No (B)	7.4-2	2.75
	684	No (B)	7.4-2	2.77

System	Problem No.	Located in Inaccessible Areas	FSAR Fig. No.	
	693	No (B)	7.4-2	2.79
	668	Yes	7.4-2	2.81
Drywell	912	No (C)	5.2-9	2.83
Inerting,				2.84
CAD & Purge				2.85
	893	No (C)	5.2-9	2.86
	894	No (C)	5.2-9	2.88
	894X	No (C)		2.90
	733	No (C)	5.2-9	2.92
	740	No (C)	5.2-9	2.94
Main Steam				2.97
	574	Yes	4.11-1	2.98

System	Problem No.	Located in Inaccessible Areas	FSAR Fig. No.	
	575	Yes	4.11-1	3.1
	631	Yes	4.11-1	3.3
	891	Yes	4.11-1	3.5
	716	Yes	4.11-1	3.7
	714	Yes	4.11-1	3.9
	715	Yes	4.11-1	3.11
	717	Yes	4.11-1	3.13
	718	Yes	4.11-1	3.15
	719	Yes	4.11-1	3.17
	720	Yes	4.11-1	3.19

System	Problem No.	Located in Inaccessible Areas	FSAR Fig. No.	
	721	Yes	4.11-1	3.21
	722	Yes	4.11-1	3.23
	723	Yes	4.11-1	3.25
	724	Yes	4.11-1	3.27
	725	Yes	4.11-1	3.29
Feedwater	578 (B1)	Yes	10.8-2	3.32
Service Water	863	No (A)	9.7-1	3.36 3.37
	865	No (A)	9.7-1	3.40
	874 (B5)	No (A)	9.7-1	3.42
	902 (B5)	No (A)	9.7-1	3.44

POOR ORIGINAL

271 089

System	Problem No.	Located in Inaccessible Areas	FSAR Fig. No.	
	901 (B5)	No (A)	9.7-1	3.46
	900	No (A)	9.7-1	3.48
	881	No (A)	9.7-1	3.50
	875	No (A)	9.7-1	3.53
	876	No (A)	9.7-1	3.57
Chilled Water (Admin- istration Building)	960	No (C)	NA	3.61 3.62 3.63 3.64 3.65
	958	No (C)	NA	3.68
	959	No (C)	NA	3.70

271 090

System	Problem No.	Located in Inaccessible Areas	FSAR Fig. No.	
	961	No (C)	NA	3.7
Fire				3.74
Protection				3.75
	916	No (C)	9.8-2	3.77
	917	No (C)	9.8-2	3.79
	918	No (C)	9.8-2	3.81
	919	No (C)	9.8-2	3.83
	920	No (C)	9.8-2	3.85
Combustion	945	No (B)	12.3-22	3.87
Air &				3.88
Exhaust				3.89
(Emergency				3.90
Diesel				3.91

271 091

System	Problem No.	Located in Inaccessible Areas	FSAR Fig. No.	
Generator)				3.92

271 092

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

ATTACHMENT 2

SUMMARY OF COMBINED LINE STRESSES

All problems were reevaluated using the SHOCK3 Code.

Stresses shown are Operating Basis Earthquake (OBE) Stresses
Design Basis Earthquake (DBE) Stresses

STATUS AS OF 6-27-79

System and Problem No.	Allowable Stress (PSI)	Reevaluation Maximum Stress	Comments
Standby Gas Treatment			
942	18000 / 27000	11805 / 11978	
941	18000 / 27000	2063 / 2684	
Control Rod Drive			
909	17220 / 25830	8258 / 10483	
Residual Heat Removal			
637	18000 / 27000	8039 / 7448	
641 (B4)	18000 / 27000	12348 / 11346	
643	18000 / 27000	12762 / 18924	

271 093

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SUMMARY OF COMBINED LINE STRESSES

System and Problem No.	Allowable Stress (PSI)	Reevaluation Maximum Stress	Comments	
646 (B6)	18000 / 27000	15442 / 13961		1.77 1.78
647	18000 / 27000	15544 / 14717		1.80 1.81
650 (B3, B6)	18000 / 27000	12309 / 7458		1.84 1.85
657	18000 / 27000	15664 / 18192		1.88 1.89
664 (B2)	18000 / 27000	10231 / 11520		1.92 1.93
682 (B2)	18000 / 27000	10734 / 9169		1.96 1.97
737 (B6)	18000 / 27000	12455 / 15081		1.99 2.1
738 (B2)	18000 / 27000	11347 / 13884		2.4 2.5
739	18000 / 27000	9169 / 9996		2.9 2.10
757 (B3, B6)	18000 / 27000	7823 / 9871		2.13 2.14
948	18000 / 27000	7982 / 8137	271 094	2.16 2.17

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SUMMARY OF COMBINED LINE STRESSES

System and Problem No.	Allowable Stress (PSI)	Reevaluation Maximum Stress	Comments	
951	18000	9107		2.19
	27000	10424		2.20
888	18000	17681		2.23
	27000	18022		2.24
867	18000	4154		2.27
	27000	4057		2.28
870	18000	4755		2.30
	27000	5663		2.31
871	18000	4416		2.34
	27000	4338		2.35
877	18000	11713		2.39
	27000	11716		2.40
879	18000	9813		2.44
	27000	8634		2.45
880 (B4)	18000	12348		2.48
	27000	11346		2.49
864	16500	12053		2.52
	24750	11393		2.53
866	18000	16123		2.56
	27000	13599		2.57
868 and 869	18000	4623		2.59
	27000	4589		2.60

271 095

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SUMMARY OF COMBINED LINE STRESSES

System and Problem No.	Allowable Stress (PSI)	Reevaluation Maximum Stress	Comments	
878	18000 27000	5137 4890		2.63 2.64
Standby Liquid Control				2.68 2.69 2.70
931	21840 32760	9273 9207		2.72 2.73
Reactor Water Cleanup				2.77 2.78 2.79
666	17390 26000	14466 17003		2.81 2.82
Reactor Core Isolation Cooling				2.86 2.87 2.88 2.89
656	18000 27000	8695 8531		2.91 2.92
667	18000 27000	3849 11716		2.95 2.96
742	18000 27000	6362 6133		2.99 3.1
933	18000 27000	15953 14885		3.4 3.5

271 096

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SUMMARY OF COMBINED LINE STRESSES

System and Problem No.	Allowable Stress (PSI)	Reevaluation Maximum Stress	Comments	
Core				3.8
Spray				3.9
651	18000	15121		3.11
	27000	22294		3.12
669 (B2, B3)	18000	6703		3.16
	27000	14637		3.17
673	18000	3735		3.20
	27000	3761		3.21
674	18000	16063		3.24
	27000	14576		3.25
934	18000	9030		3.29
	27000	9687		3.30
Reactor				3.34
Building				3.35
Cooling				3.36
Water				3.37
873	18000	7686		3.39
	27000	10768		3.40
872	18000	8030		3.44
	27000	11121		3.45

271 097

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ATTACHMENT 2

SUMMARY OF COMBINED LINE STRESSES

System and Problem No.	Allowable Stress (PSI)	Reevaluation Maximum Stress	Comments	
Fuel Pool Cooling & Cleanup				3.49 3.50 3.51
949	21960 32940	10425 10488		3.53 3.54
950	21180 31770	19548 20362		3.57 3.58
952	18000 27000	14557 13340		3.60 3.61
953	18000 27000	4057 6291		3.64 3.65
947	18000/20850 27000/31275	4100/18982 3870/17379		3.68 3.69
High Pressure Coolant Injection				3.72 3.73 3.74 3.75
655	18000 27000	9006 9613		3.77 3.78
679 (B1)	18000 27000	16955 16691		3.82 3.83
681	18000 27000	8506 8229		3.86 3.87

271 098

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SUMMARY OF COMBINED LINE STRESSES

System and Problem No.	Allowable Stress (PSI)	Reevaluation Maximum Stress	Comments	
684	18000/20652	8481/8202		3.90
	27000/30978	8179/8502		3.91
693	22500	22485		3.94
	33750	15245		3.95
668	18000	3993		3.98
	27000	7913		3.99
Drywell Inerting CAD & Purge				4.4
				4.5
				4.6
				4.7
912	18000	8233		4.9
	27000	10880		4.10
893	18000	10328		4.14
	27000	10062		4.15
894	18000	8616		4.19
	27000	22787		4.20
894x	18000	15035		4.22
	27000	26364		4.23
733	18000	9526		4.26
	27000	10977		4.27
740	18000	5498		4.29
	27000	6599		4.30

271 099

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SUMMARY OF COMBINED LINE STRESSES

System and Problem No.	Allowable Stress (PSI)	Reevaluation Maximum Stress	Comments	
Main Steam				4.33 4.34
574	18000 27000	10096 11078		4.36 4.37
575	18000 27000	13026 15624		4.41 4.42
631	18000 27000	12941 15704		4.46 4.47
891	18000 27000	15754 19260		4.50 4.51
716	18000 27000	7773 8346		4.54 4.55
714	18000 27000	7086 7126		4.59 4.60
715	18000 27000	12445 10589		4.64 4.65
717	18000 27000	6892 8414		4.69 4.70
718	18000 27000	8153 7191	271 100	4.74 4.75
719	18000 27000	5986 5829		4.79 4.80

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ATTACHMENT 2

SUMMARY OF COMBINED LINE STRESSES

System and Problem No.	Allowable Stress (PSI)	Reevaluation Maximum Stress	Comments	
720	18000 / 27000	3706 / 4794		4.84 4.85
721	18000 / 27000	16931 / 14278		4.89 4.90
722	18000 / 27000	11207 / 9619		4.94 4.95
723	18000 / 27000	11754 / 11852		4.99 5.1
724	18000 / 27000	2622 / 2785		5.5 5.6
725	18000 / 27000	3808 / 3806		5.9 5.10
Feedwater				5.12
578 (B1)	18000 / 27000	16874 / 20378		5.15 5.16
Service Water				5.20 5.21
863	16500 / 24750	6156 / 5954		5.23 5.24
865	18000 / 27000	9171 / 8359		5.28 5.29
874 (B5)	18000 / 27000	3920 / 3845	271 101	5.32 5.33

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

ATTACHMENT 2

SUMMARY OF COMBINED LINE STRESSES

System and Problem No.	Allowable Stress (PSI)	Reevaluation Maximum Stress	Comments	
902 (B5)	18000 / 27000	3682 / 3197		5.36 5.37
901 (B5)	18000 / 27000	4122 / 3826		5.41 5.42
900	18000 / 27000	3188 / 2993		5.46 5.47
881	18000 / 27000	17466 / 23135		5.51 5.52
875	18000 / 27000	10376 / 10417		5.56 5.57
876	18000 / 27000	5345 / 4631		5.61 5.62
Chilled Water (Administration Bldg.)				5.66 5.67 5.68 5.69
960	18000 / 27000	2017 / 2284		5.71 5.72
958	18000 / 27000	6913 / 6856		5.75 5.76
959	18000 / 27000	986 / 1056	271 102	5.79 5.80
961	18000 / 27000	1207 / 1411		5.84 5.85

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

ATTACHMENT 2

SUMMARY OF COMBINED LINE STRESSES

System and Problem No.	Allowable Stress (PSI)	Reevaluation Maximum Stress	Comments	
Fire				5.89
Protection				5.90
916	18000	7527		5.92
	27000	7335		5.93
917	18000	3060		5.96
	27000	3118		5.97
918	18000	3681		6.1
	27000	3711		6.2
919	18000	6659		6.6
	27000	6414		6.7
920	18000	3827		6.11
	27000	3823		6.12
Combustion				6.16
Air & Exhaust				6.17
Emergency				6.18
Diesel Gen.				6.19
945	18000	5212		6.21
	27000	4520		6.22

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ATTACHMENT NOT FILMED

ANO. 7907020270

NO. OF PAGES 17

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ATTACHMENT 5

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
STRESS RE-EVALUATION
CONSERVATIONS; CLARIFICATIONS AND ADDITIONS

The following provides additions to the conservatisms noted in the letter of June 8, 1979 (JPN-79-32) from Mr. P. J. Early of the Power Authority of the State of New York to Mr. T. A. Ippolito of the Nuclear Regulatory Commission's Division of Operating Reactors:

1. Combination of Stresses

   The stress values for each loading condition are combined in accordance with the intent of ANSI B31.1.0-1967 to determine the acceptability of piping stress. Since the procedure used stresses from individual loading cases, and does not combine moments as per ASME Code III, the calculated total stresses are more conservative than the stresses calculated with the present ASME Code III. (Clarifies and supercedes Conservatism No. 1, first paragraph, of JPN-79-32).

2. Constraint Load Combinations

   The evaluation of constraint design considered the coincident application of deadload, temperature and occasional loads, including earthquake, even though their occurrence may not be simultaneous. Consideration of realistic application of these loads would result in reduced loadings. (Clarifies and supercedes Conservatism No. 1, second paragraph, JPN-79-32).

3. Floor Response Spectra

   Seismic response for these conservatively postulated earthquakes is represented by families of ground response spectra which envelope the effects of ground motion upon a suitable range of damped single-degree-of freedom (SDF) oscillator systems. To determine structural response to seismic loadings a mathematical model is developed which closely approximates the real structural containment system in physical and response characteristics. Amplified response spectra are generated from the structural response for specific floors (elevations) in the structure. This is accomplished by using a damped sinusoidal support motion (i.e. modal response of structure to ground-shock-spectral) to determine the response of a range of damped SDF oscillators. The procedure is carried out for all significant modes of response of the structural system. (Supercedes Conservatism No. 3, JPN-79-32).

271 105

4. The analyses and reanalyses of seismic piping systems are based upon the conservative stress limit of 1.8Sh under DBE loading conditions. The corresponding ASME Section III Code piping stress limit is 2.4Sh under the DBE conditions. In July 1978, the NUREG/CR-0261 report, using the limit moment theory to address the Code rules, established that gross plastic deformation may occur when primary stress exceeds 1.5 to 2.0 times the yield

strength (Sy) of piping material which corresponds to 2.4 to 3.2 Sh or higher.

5. Computed pipe stresses are magnified by the application of ANSE B31.1. 0-1967 intensification factors, which we believe were intended to represent fatigue factors and thus are not strictly applicable to the seismic load conditions.

ATTACHMENT 6

HIGH ENERGY PIPE BREAKS

The analysis of high energy pipe breaks, inside and outside the primary containment, are discussed, respectively, in FSAR Apppendix E and the Special Report on the Effects of a High Energy Piping System Break Outside the Primary Containment (Supplement 25 to the FSAR).

The original analysis inside the primary containment considered, at least, the ten highest stress points on each of the main, steam, feedwater, core spray, HPCI and RCIC lines. As these points encompass the high stress points determined during the reevaluation effort, the pipe break analysis presented in Appendix E remains valid.

As described in FSAR, all safety related, high energy lines outside the primary containment were originally considered and analyzed for pipe break as ASME Code III systems. For each line, terminal points plus two intermediate points were analyzed for pipe breaks. The terminal points are fixed. The reevaluation stresses at the intermediate points were reviewed and none were found to exceed the 0.8 (S_h+S_a) criteria. Therefore, the criteria used originally for the selection of postulated break points, based on possible impacts, and the points selected, remain valid.

In addition, all high energy piping penetrations of the primary containment have been reevaluated and found to be acceptable.

Therefore the original high energy pipe break analysis remains valid, and such validity has not been affected by the current pipe stress reevaluation.

271 107