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Docket No. 50-333

AUGUST 14 1979

Mr. George T. Berry
General Manager and
Chief Engineer
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

The Commission today has issued the enclosed Order lifting the suspension of facility operation required by the Order to Show Cause dated March 13, 1979, for the James A. FitzPatrick Nuclear Power Plant. The enclosed Order also confirms and requires certain commitments made by the Power Authority of the State of New York including a commitment to complete re-analysis of piping supports in accessible areas within 60 days of the date of plant startup.

This Order is being issued because your reanalysis and modifications of piping deficiencies in safety related systems, along with the constraints during the 60 day period required by the Order, have demonstrated that FitzPatrick can safely withstand the effects of seismic events should they occur. The basis for this action is set forth in the Order.

Sincerely,

Original signed by

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

Enclosure:
Order

cc w/enclosure:
See next page

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cc:

Mr. Vito J. Cassan
Assistant General Counsel
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Mr. Peter W. Lyon
Manager - Nuclear Operations
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Mr. J. D. Leonard, Jr.
Resident Manager
James A. FitzPatrick Nuclear
Power Plant
P. O. Box 41
Lycoming, New York 13093

Director, Technical Development
Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Oswego County Office Building
46 E. Bridge Street
Oswego, New York 13126

George M. Wilverding, Licensing Supervisor
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the matter of	)	
	)	
The Power Authority of the State of New York (James A. FitzPatrick Nuclear Power Plant)	)	Docket No. 50-333

ORDER

The Power Authority of the State of New York (the licensee) is the holder of Facility Operating License No. DPR-59 which authorizes operation of the James A. FitzPatrick Nuclear Power Plant at power levels up to 2436 megawatts thermal (rated power). The facility, which is located at the Licensee's site in Oswego County, New York, is a boiling water reactor (BWR) used for the commercial generation of electricity.

II.

Because certain safety related piping systems at the facility had been designed and analyzed with a computer code which summed earthquake loads algebraically, the potential existed for compromising the basic defense-in-depth provided by redundant safety systems in the event of an earthquake. This potential compromising resulted from the possibility that an earthquake of the type for which the plant must be designed could cause a pipe rupture as well as degrade the emergency cooling system designed to mitigate such an accident. Therefore, by Order of the Director of Nuclear Reactor Regulation (the Director) for the Nuclear Regulatory Commission (NRC), dated March 13, 1979 (44 FR 16511, March 19, 1979), the licensee was ordered to show cause:

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(1) Why the licensee should not reanalyze the facility piping systems for seismic loads on all potentially affected safety systems using an appropriate piping analysis computer code which does not combine loads algebraically;

(2) Why the licensee should not make any modifications to the facility piping systems indicated by such reanalysis to be necessary; and

(3) Why facility operation should not be suspended pending such reanalysis and completion of any required modifications.

In view of the importance of safety of this matter, the Order was made immediately effective and the facility was required to be placed in the cold shutdown condition and remain in that mode until further Order of the Commission.

III.

The facility is currently in the cold shutdown condition. Pursuant to the March 13, 1979 Order, the licensee filed a written answer to the Order by letter dated March 30, 1979 (date of receipt). In this response the licensee stated that it is reanalyzing all potentially affected safety systems for seismic loads using an appropriate method which does not sum loads algebraically.

By letter dated August 2, 1979, the licensee requested the startup of the James A. FitzPatrick Nuclear Power Plant. This request is based on: (1) the completion of the analysis for all piping, equipment nozzles, and containment penetrations for both the Design Basis Earthquake (DBE) and Operating Basis Earthquake (OBE), (2) the completion of all analyses and modifications for those pipe supports in areas inaccessible during normal plant operation, (3) the completion of modifications identified to date to those pipe supports in areas accessible during normal plant operation, and (4) a commitment to complete the analysis of the remaining pipe supports in accessible areas within 60 days from the date of plant startup. Technical Support for these conclusions is provided in letters from the licensee dated March 30, (date of receipt), June 8, 28, and August 2, 7, 10, 14, 1979, and letters from Stone and Webster dated March 22, 30, April 3, 6, 11, 13, 18, 27 and May 11, 14, 18, 1979. The licensee has committed: (1) to shutdown the facility if a seismic event occurs, which results in accelerations greater than an acceleration level of 0.01 g, the setpoint of the facility's accelerometers, and (2) in the event of a 0.01 g seismic event to inspect those piping systems and supports which have not been shown to be fully acceptable for the Operating Basis Earthquake (OBE) case (ground acceleration of 0.07 g). This commitment is required only until such time that the reanlaysis for the OBE loading condition, and any necessary modifications, is completed. In addition, the licensee has committed to notify the NRC within twenty four (24) hours if it is determined that any of the remaining support analyses result in declaring a support inoperable. Based on the above, the licensee contends that good cause has been shown why the suspension of facility operation should not be continued in effect while the reanalyses of the remaining pipe supports are completed.

The licensee's analyses were performed using the PSTRESS SHOCK 3 computer code which combines stresses in a manner acceptable to the NRC staff. The reanalyses resulted in the calculations of some stresses above allowable. In these cases, when the calculated stresses on piping indicated that support loadings were above original design values, the licensee was required to reanalyze the suppport.

The licensee reanalyzed 96 pipe stress problems as a result of the March 13, 1979 Show Cause Order. Five problems required hardware modifications. Of these 5 problems, one required modifications to supports as a result of seismic overstresses. The other four modifications were required because of verification of "as-built" conditions, thermal stresses, and modeling differences. Of a total of 989 supports the licensee has evaluated all 335 pipe supports in areas inaccessible during normal plant operation as well as 273 supports in areas accessible during plant operation. Of these 608 analyzed existing supports, 29 required modifications, with a few of these modifications due to significant load increases. Nine other modifications in the form of installation of new additional supports resulted from "as-built" conditions.

The NRC staff has reviewed the licensee's submittals. This review included, among other things, an evaluation of the codes which compute pipe stresses resulting from the facility's response to an earthquake. The means by which piping responses are combined in the codes that are currently a basis for the facility design are summarized below:

PSTRESS/SHOCK 3

This code combines intramodal* responses by a modified the square root of the sum of the squares (SRSS) and combines intermodal* responses by SRSS or absolute sum for closely spaced modes.

The NRC staff has determined that an algebraic summation of responses was not incorporated into the PSTRESS/SHOCK 3 code. The NRC staff has further concluded that this code provides an acceptable basis for analyzing the facility piping design.

Based on the NRC Staff's Safety Evaluation dated August 14 , 1979, the staff finds that all piping, equipment nozzles, and containment penetrations affected by the March 13, 1979 Show Cause Order and all piping supports located in areas inaccessable during normal plant operation and approximately one half of those located in accessable areas have been acceptably reanlayzed and modified and/or repaired as necessary.

The remaining 381 pipe supports in areas accessible during normal plant operation will be analyzed within sixty (60) days of plant startup. Based on the results of the analysis of supports in areas inaccessible during normal plant operation (i.e., as of August 6, 1979, 1 of 335 have a safety factor of less than 2 with respect to ultimate capacity as described in the safety evaluation), it is expected that very few, if any, supports in accessible areas have a safety factor of less than 2 with respect to ultimate capacity.

*Modes are defined as dynamic piping deflections at a given frequency. Intramodal responses are the components of force, moment and deflecton within a mode. Intermodal responses are the components of force, moment and deflection of all modes.

The remaining supports in accessible areas are on systems which are less critical to safe shutdown than those in inaccessible areas. There is no increased potential for a loss-of-coolant accident because the reactor coolant pressure boundary is in an inaccessible area and has been reanalysed. In addition, the analysis of the remaining accessible supports, and modifications to insure system operability if necessary, will be completed within sixty (60) days of startup and an earthquake approaching the DBE in this time period is very unlikely. The licensee has provided a schedule for completion of remaining support analyses. This schedule results in completion of at least one train of all redundant safety systems within 30 days. In the event a support is found to be above design load, a determination will be made of the significance of the load, and modifications will be made. Those supports that fall in this category may, depending on the load level, be declared inoperable as defined in the Technical Specifications.

The licensee to date has not completed the actions identified in paragraph number 2 of the Order to Show Cause dated March 13, 1979 and this Order does not affect that portion of the March 13, 1979 Order. The licensee has, pursuant to paragraph 3 of the Order, shown cause why operation of the facility should not remain suspended pending the completion of reanalyses and completion of any further required modifications.

The licensee's answer to the Order did not request a hearing nor did any person request a hearing.

IV.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Parts 2 and 50, IT IS DETERMINED THAT: The public health, interest or safety does not require the continued shutdown of the facility, AND IT IS HEREBY ORDERED THAT:

1. Effective this date the suspension of facility operation required by the Order to Show Cause of March 13, 1979 is lifted.

2. All modifications to correct all piping systems, equipment nozzles and containment penetrations and all modifications to supports located in areas inaccessable during normal plant operation shall be completed prior to plant startup.

3. The licensee shall both complete reanalyses of the remaining pipe supports in areas accessible during normal plant operation and propose a schedule for implementation of any necessary modification within sixty (60) days of plant startup.

4. For each modification identified as a result of reanalysis of the remaining supports in accessible areas after resumption of facility operation, when a support is deemed inoperable (a support will be considered inoperable if the loads exceed a factor of safety of 2 to ultimate and exceed the limits of ASME B&PV Code, Section III, Subsection NF) the NRC shall be notified within 24 hours after making each such determination. The affected system shall be considered inoperable until the necessary modifications are implemented within seven days or the time frame allowed by the facility Technical Specifications, whichever is less, unless a reanlaysis of the affected piping system is performed which: (1) demonstrates that the overstressed support remains operable, or (2) demonstrates, with the overstressed support removed from the system, that the system remains operable.

5. The James A. FitzPatrick Nuclear Power Plant shall be shutdown if an earthquake with an acceleration greater than .01 occurs (site accelerometers are set of 0.01 g) and the licensee shall inspect all safety-related piping systems which have not been reanalyzed and shown to be acceptable at the 0.07 g level of the OBE. Prior to resuming operations following an earthquake, the licensee shall demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public.

FOR ~~THE~~ NUCLEAR REGULATORY COMMISSION

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 14 day of August, 1979

SAFETY EVALUATION BY THE OFFICE OF

NUCLEAR REACTOR REGULATION

FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

Introduction

On March 13, 1979, the Commission issued an Order to Show Cause to the Power Authority of the State of New York (licensee) requiring that James A. FitzPatrick Nuclear Power Plant (facility) be placed in cold shutdown and the licensee show cause:

(1) Why the licensee should not reanalyze the facility pipine systems for seismic loads on all potentially affected safety systems using an appropriate piping analysis computer code which does not combine loads algebraically;

(2) Why the licensee should not make any modifications to the facility piping systems indicated by such reanalysis to be necessary; and

(3) Why facility operation should not be suspended pending such reanalysis and completion of any required modifications.

The licensee's response to the Order, dated March 30, 1979 (date of receipt) stated that it is reanalyzing all potentially affected safety systems for seismic loads using an appropriate piping analysis method. The licensee also requested that the Order be modified or rescinded such that the FitzPatrick Plant would be allowed to immediately resume full power operation pending resolution of items set forth in said response.

Discussion

The Stone and Webster (S&W) PSTRESS/SHOCK 2 computer code for pipe stress analyses sums earthquake loadings algebraically and is unacceptable for reasons set forth in the March 13, 1979 Order to Show Cause. This code was used in the seismic analyses of certain safety and nonsafety related systems at the facility. The licensee has identified the seismically analyzed (Seismic Category I) systems at the facility including those analyzed with SHOCK 2. It has also identified the other methods of seismic analysis used for other Seismic Category I systems. Furthermore, the licensee has reported the results of the reanalyses of SHOCK 2 safety systems and has provided support for the acceptability of the analysis methods used on the remaining Seismic Category I systems.

We have evaluated the results of all the methods of pipe stress analysis previously utilized and used in the reanalyses for the facility.

Evaluation

1. Systems

Portions of the following systems were identified by the licensee as having been analyzed with SHOCK 2:

1. Standby Gas Treatment
2. Control Rod Drive
3. Residual Heat Removal
4. Standby Liquid Control
5. Reactor Water Cleanup
6. Reactor Core Isolation Cooling
7. Core Spray
8. Reactor Building Cooling Water
9. Fuel Pool Cooling and Cleanup
10. High Pressure Coolant Injection
11. Drywell Inerting, CAD and Purge
12. Main Steam
13. Feedwater
14. Service Water
15. Chilled Water
16. Fire Protection
17. Combustion Air and Exhaust Emergency Diesel Generator

The licensee has reanalyzed 96 pipe stress problems originally analyzed by SHOCK 2. All supports in areas inaccessable during normal plant operation, including areas inside containment, were reanalyzed and modifications will be completed prior to startup. All of the analyses completed have included both the Operating Basis Earthquake (OBE) and Design Basis Earthquake (DBE) loadings. A portion of the supports outside containment have been analyzed and the remainder will be reanalyzed within sixty (60) days of the date of plant startup.

Ninety one of the stress problems were determined to have pipe stress values after reanalysis, considering "as-built" conditions, within acceptable allowable values.

The remaining 5 problems were resolved as follows:

(1) Problem 733 (A) (Drywell Vent and Purge) - The "as-built" inspection disclosed that a reinforcing pad on a 30" x 20" Tee had been omitted during construction. By modification of the support H27-4, the stresses were reduced to within allowable for the unreinforced Tee.

(2) Problem 650 (Residual Heat Removal) - Initial reanalysis of the problem showed the pipe stress to be acceptable. When this problem was remodeled and reanalyzed including all appropriate branches, one TEE was over-stressed when the stress intensification factor was considered. Two snubbers H10-50N and H10-51CN were added to reduce the stress to within allowable.

(3) Problem 657 (RHR-Head Spray) - Field verification disclosed that a valve was located about three feet and one 90 degree bend from its original analyzed position. This resulted in exceeding pipe stress allowables by 25% for the OBE case. Pipe stress was satisfactory for the DBE case. Snubbers H10-383N and H10-387 were added to resolve this problem.

(4) Problem 947 (A)(Fuel Pool Cooling) - Reanalysis indicated an 80% over stress existed for the OBE case. Pipe stress was satisfactory for the DBE case. Supports H19-215N and FPSK 1000N were added to resolve this problem.

(5) Problem 909 (Control Rod Drive Cooling) - The interface between the Category I and non-safety related piping was independently defined. To simplify the analysis, Suport Q8-160 (terminal anchor) was added. This was not a pipe stress problem but one of ensuring complete documentation of safety related stress analyses.

2. Verification of Analysis Methods

We have reviewed the acceptability of the analytical methods which are currently a basis for the facility piping design. The licensee has identified the following computer codes/analysis methods as applicable:

PSTRESS/SHOCK 3
Static Analysis Methods

PSTRESS/SHOCK 3

S&W has stated that PSTRESS/SHOCK 3 calculates the intramodal responses by adding the absolute value of the response due to the vertical earthquake excitation to the (SRSS) combination of the response due to the two horizontal earthquake components. The intermodal components are calculated by the SRSS method. A review of the code listing has confirmed these statements.

S&W has also solved three benchmark piping problems provided by the NRC with this code, and its solutions show acceptable agreement with the benchmark solutions. In addition, a comparison of the S&W and BNL solutions of the confirmatory problem also demonstrate good agreements (within 10%).

Static Analysis

Much of the 6 inch and smaller Category I piping at FitzPatrick was analyzed using simplified static methods. The methods were intended to keep the fundamental piping frequencies out of the range of the fundamental structural frequency by establishing span lengths between supports. Calculations were based on simple beam formulations. Tabulations relating various spans, nominal pipe sizes, and acceleration levels to actual pipe stress levels were provided for use by the analyst. The acceleration applied to the piping was dependent upon where the piping fundamental frequency was relative to the structural frequency. Calculated seismic stress was based on an assumed three component earthquake. Support loadings were based on standardized loadings enveloping the various loading conditions. Nozzle loads were calculated based on similar, simplified methods.

Piping two inches and below was shown on the piping drawings "diagrammatically" (i.e., without detailed dimensions). The stress engineers located supports during the installation process working at the site with erection isometric sketches.

3. Reanalysis Methods and Results

The safety related piping systems at the FitzPatrick Nuclear Plant have been reviewed to determine the method of analyses. Ninety six (96) computer stress problems of safety related piping have been identified where the analysis used the computer code SHOCK 2 which used an algebraic intramodal summation of responses to earthquake loadings. The problems where an algrbraic intramodal response combination technique was used in the design have been reevaluated using acceptable methods. The reevaluation included a dynamic computer analysis using SHOCK 3, which incorporated a lumped mass response spectra modal analysis technique.

The floor response spectra used in the reanalysis was the original amplified response spectra specified in the FSAR. The peaks in the amplified floor response spectra were broadened by ±15% in acordance with Regulatory Guide 1.122 to account for variation in material properties and approximations in modeling.

The piping systems were modeled as three dimensional lumped mass systems which included consideration of eccentric masses at valves and appropriate flexibility and stress intensification factors. The dynamic analysis procedures meet the criteria specified in the plant FSAR and are acceptable. The resultant stresses and loads from the reanalysis were used to evaluate piping, supports, nozzles, and penetrations.

All of the 96 SHOCK 2 pipe stress problems have been reanalyzed and verified by Stone and Webster Engineering Assurance and the licensee's Quality Assurance Program. This reanalysis completed the entire scope of piping stress reanalysis. Based on our review of the computer codes being used for reanalysis, independent check analysis performed by the staff and a review of modeling methods used by the licensee, we find acceptable the procedures and methods used in reanalyzing these problems.

I&E Bulletin 79-04, "Velan Valve Weights", has been addressed and resolved for all 96 piping reanalyses.

At the request of the NRC, its consultant, EG&G INEL performed audit pipe stress calculations of five FitzPatrick problems using the NUPIPE computer code. The results of the EG&G audit compare favorably with the results of the licensee's results.

The pipeline support designs for affected system piping was inspected by the licensee to verify the location, orientation, support clearances, and support type. Any deviations that were identified are incorporated into piping reanalyses. These piping systems were also verified by the NRC Office of Inspection and Enforcement.

The pipe supports were reevaluated in cases where the original suppport design loading was exceeded as a result of piping reanlaysis. In cases where the original support capacity was exceeded, the support reevaluation has included the consideration of base plate flexibility and a verification of actual field construction of the support. Where concrete expansion anchor bolts were used, their capacities, without compromising the originally committed safety margin, were also included in the reevaluation.

There are 989 supports on lines originally analyzed by SHOCK 2; of these all of the suports 335 in inaccessible areas including inside containment, and 273 of the supports in accessible areas have been evaluated and all necessary modifications will be completed prior to operation of the facility. Nine new supports were added to the piping systems, and 29 of the existing 608 supports analyzed to date were identified to require modifications. There are approximately 381 supports remaining to be evaluated. During the reanalysis it was determined that the majority of the support modifications arose as a result of the "as-built" supports which deviated from the original design. Only one of them can be qualified as due to inadequate, original seismic analysis incorporating algebraic summation technique.

One support in inaccessible areas would not meet the intent of a factor of safety of 2. This support was a "U" bolt originally installed for a lateral constraint. A lateral overstress of 339% was determined. The configuration of this support is such that deformation would occur but it would remain functional. The anchor bolts are also 104% overstressed but still have a safety factor of 2.5.

Based on the results to date, we expect other supports may be found that will be above allowable limits. In the event the loads on a pipe support exceed allowable loads, the support will be considered operable if its loads do not exceed the limits of ASME B&PV Code, Section III, Subsection NF, or a factor of safety of 2 to ultimate. If support reanalysis indicates that a support is inoperable, we have required the licensee to inform the NRC of the results of reanlaysis within 24 hours and that the affected system be considered inoperable as specified in the facility Tecnnical Specifications until the necessary modifications are implemented or a reanalysis assuming support failure is completed.

Five supports in accessable areas exceed conservative local stress limits, at attachment welds. Modification of these supports is being made to satisfy these limits. It should be recognized that such local stresses are calculated based on theoretical elastic response. Further, these limits are purposely set quite low to enable the weld to withstand the fatigue cycles which will be endured throughout the 40 year lifetime of the plant. JAFNP has been in operation for less than four years; therefore, considerable reserve margin exists. This is particularly true since these fatigue allowables normally contain a factor of safety of 20. Should a DBE occur with its low number of stress cycles, the attachment welds would continue to function satisfactorily. The only concern would be a shortening of the weld's fatigue life which in no way would affect its ability to withstand an earthquake. The performance of the overall support structure should remain elastic. It should be noted that these supports are acceptable under the criteria to which they were designed.

Loads of attached equipment nozzles and penetrations were checked and verified to be either below the initial allowable values or were evaluated and determined to be acceptable. Confirmation of the results of reanlaysis have been obtained from the equipment manufacturers where necessary.

The design and analysis of the supports and attached equipment are in accordance with the criteria specified in the plant FSAR.

The pipe break criteria of the FSAR was reviewed in connection with the possible effect of changes of the high stress point resulting from the reanalyses. The piping systems and supports were designed to the allowable limits on ANSI B31.1 for the gross properties and to the limits of AISC structure steel code edition six. Results of the evaluation of the effect the reanalyses has on the FSAR pipe break criteria show that no new whip restraints are required. Therefore, we find that the reanlaysis has not changed the pipe break protection.

The safety related piping systems supports and attached equipment, where the original analysis used an algebraic intramodal summation technique, have been, or are to be reanlayzed with acceptable methods. The procedures used in the support reanalyses and their results have been reviewed against the criteria in the FSAR and found acceptable.

4. Conclusion

The licensee has demonstrated that SHOCK 2 is the only method of analysis used for the facility's safety related systems which combines seismic loads algebraically. Safety related piping systems analyzed with SHOCK 2 have been reanalyzed with an acceptable dynamic code. Results of the reanalysis indicated that the pipe stress and equipment loads, after necessary modifications, will be acceptable when compared with the FSAR allowables and the manufacturer's specified load criteria.

The reevaluation of pipe stress problems indicated that modifications in three problems were necessary as a result of the seismic reanalysis. One problem was modified due to an "as-built" condition which resulted in piping overstress. These modifications are identified in Section 1. The licensee will complete all modifications inside containment prior to plant operation. Evaluation of the supports and schedule for completion of necessary modifications outside of containment will be completed within sixty (60) days of the date of the Order. Further, in those cases where reanalysis exceeds code allowable, the staff requires that the criteria used to determine whether a factor of safety of 2 to ultimate does exist be linear elastic analysis techniques or no more than twice the rated load for snubbers. Use of detailed finite element analysis for evaluation of local stresses due to integral attachment is acceptable. Supports in accessable areas which exceed the factor safety of two to ultimate or the limits of ASME B&PV Code Section III, Subsection NF will be considered as inoperable as defined in the Technical Specifications.

We reviewed the analysis techniques which are currently the bases for the facility's piping design. We have determined that the application of these techniques at FitzPatrick assures that safety related systems will withstand the design basis earthquake. Although the reanalysis of supports outside containment is not complete, there is reasonable assurance that the facility can operate during the interim period until the reanalysis and any required modifications are completed without endangering the health and safety of the public. This assurance is based on the following factors:

(1) All safety system piping both inside and outside containment which was originally seismically analyzed with the SHOCK 2 program has been reanalyzed and, subject to modification, is, or was made, acceptable.

(2) All of the affected safety systems inside containment have been reanlayzed (piping, supports, nozzles, and penetrations) and were found either acceptable as presently designed or will be modified as identified in this SER prior to startup. Modifications which still remain are delineated in attachement 2 of the licensee's submittal of August 2, 1979.

(3) The review of 608 supports identified 1 support that would not meet the intent of a safety factor of 2. It is therefore, reasonable to expect that few remaining supports would exceed a safety factor of 2.

(4) Confirmation of input data through "as-built" verification provides assurance that analytical results are correct and significant "as-built" deficiencies repaired.

(5) The licensee has completed the analysis for the High Pressure Core Injection and Reactor Core Isolation Cooling Systems assuring that these ECCS systems and systems necessary for maintaining hot shotdown will be capable of withstanding a design basis earthquake.

(6) The licensee has committed to complete all the support reanalysis in accessable areas outside containment within sixty (60) days of the date of plant startup.

(7) The probability of an earthquake exceeding the design basis earthquake during the sixty (60) day period (or the thirty (30) day period for finalizing the reanalysis for one of the two redundant trains for each safety system) that the remaining support analysis is being completed is small and the licensee has committed to shut down the facility in the event of an earthquake which exceeds 0.01 g acceleration and inspect all piping, penetrations, supports and nozzles which have not been reanlayzed for both OBE and DBE.

(8) The NRC will require prompt notification of inoperable supports within twenty four (24) hours and either resolution by reanalysis of the piping system assuming a failed support or modification of the affected support, if reanalysis of a support exceeds the factor of safety of two to ultimate and the limits of ASME B&PV Code, Section III, Subsection NF.

Based on the above, we conclude that the licensee has shown cause why FitzPatrick can be operated for 60 days pending completion of reanalyses required by the Show Cause Order of March 13, 1979.

Dated: August 14, 1979