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:

357196

7909170640

(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.

957197

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.

957198

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

957203