UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of)
)
Virginia Electric and Power Company) Docket No. 50-280
(Surry Power Station, Unit No. 1))

ORDER

I.

The Virginia Electric and Power Company (the licensee) is the holder of Facility Operating License No. DPR-32 which authorizes operation of the Surry Power Station, Unit No. 1 at power levels up to 2441 megawatts thermal (rated power). The facility, which is located at the licensee's site in Surry County, Virginia, is a pressurized water reactor 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 to 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 April 2, 1979. 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 1, 1979, the licensee requested the startup of Surry Power Station, Unit 1. This request is based on the completion of all pipe stress analyses for the Design Basis Earthquake (DBE), the completion of all analyses for those pipe supports inside containment for the DBE, the completion of all modifications to the supports inside containment, and a commitment to complete the analyses of pipe supports outside containment within 60 days from the date of this Order. Technical Support for these conclusions is provided in the "Report on the Reanalysis of Safety-Related Piping Systems, Surry Power Station, Unit 1" dated June 5, 1979 and letters from the licensee dated March 30, April 23, 24, 27, May 2, 22, 24, 30, June 4, 8, 12, 15, 19, 25, and August 1, 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 to (1) shut down 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) 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 reanalysis for the OBE loading condition, and any necessary modifications, is completed. 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 NUPIPE computer code, which combines stresses in a manner acceptable to the NRC staff. The reanalyses resulted in the calculation of some stresses above allowable. In these cases, the licensee recalculated the stresses using soil structure interaction (SSI) methodology with a 50 percent increase in the inertia forces which the staff required to be applied to each pipe run after computer calculation of stress and support loads. This methodology with a 50 percent increase was approved by the NRC staff in its letter dated May 25, 1979. In those cases when stresses on the piping from the calculations using SSI indicated that support loadings were above original design values, the licensee was required to reanalyze the support.

The licensee reanalyzed 63 pipe stress problems which required reanalysis as a result of the March 13, 1979 Show Cause Order. Nineteen problems required hardware modifications. Of these 19 problems, four required modifications to supports as a result of seismic overstresses. Other modifications were required because of verification of "as-built" conditions, thermal stresses, and modeling differences. The licensee has also evaluated 487 pipe supports inside containment. Of the supports, 66 required modifications, and only a few of these modifications were because of significant load increases. The other modifications 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:

NUPIPE

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 NUPIPE 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 ___, 1979, the staff finds the piping affected by the March 13, 1979 Show Cause Order and all piping supports inside containment have been acceptably reanalyzed.

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

Out of a total of 518 supports outside containment, 80 had been reanalyzed as of July 21, 1979. Of these 80 supports, 5 required modification.

The remaining pipe supports outside containment will be analyzed and any modifications identified within sixty (60) days of startup. Based on the results of the analysis of supports inside containment (i.e., as of July 31, 1979, 4 of 487 have a safety factor of less than 2 with respect to ultimate capacity), it is expected that very few, if any, supports outside containment have a safety factor of less than 2 with respect to ultimate capacity. Of the 438 supports outside containment which remain to be completed, 46 are associated with high and low head safety injection, containment and recirculation spray, and auxiliary feedwater systems. The licensee has agreed to complete these priority supports prior to startup.

The remaining supports outside containment are on systems which are less critical to safe shutdown than those inside containment such as the component cooling water system. There is no potential for a loss-of-coolant accident because the reactor coolant pressure boundary is inside containment. In addition, the modifications will be completed within sixty (60) days of startup and an earthquake approaching the DBE in this time period is very unlikely. 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 other 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 piping system overstress and all modifications to supports inside containment and those identified in Table 4.1.B of the licensee's August 1, 1979 submittal shall be completed prior to startup.

3. The licensee shall complete analysis and any necessary modifications for the 46 remaining supports associated with high and low head safety injection, containment and recirculation spray and auxiliary feedwater systems prior to startup.

4. The licensee shall both complete reanalysis of the remaining pipe supports outside containment and propose a schedule for implementation of all identified modifications within sixty (60) days of the date of this Order.

5. For each modification identified as a result of reanalysis of the supports outside containment after resumption of facility operation, when the overall margin of safety of the support to ultimate capacity is determined to be less than 2, the NRC shall be notified within 24 hours after making each such determination. The affected system shall be considered inoperable as that term is used in the facility Technical Specifications until the necessary modifications are implemented within the time frame allowed by the facility Technical Specifications

unless a reanalysis of the affected piping system is performed with the overstressed support removed from the system to demonstrate that the system is operable.

6. The Surry Power Station Unit No. 1 shall be shutdown if an earthquake with an acceleration greater than 0.01 g occurs (site accelerometers exceed 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

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this day of