Docket File

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MARCH 26 1980

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Docket No. 50-281

Mr. W. L. Proffitt
Senior Vice President - Power
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Proffitt:

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 Surry Power Station, Unit No. 2.

This Order is issued because your reanalysis and modifications of piping deficiencies in safety related systems, along with the operational control required by the Order, have demonstrated that the Unit No. 2 can safely withstand the effects of seismic events should they occur in the area and because the modifications will be complete before startup. The basis for this action is set forth in the Order.

Sincerely,

for Original Signed By
E. G. Case

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Order

cc: w/enclosure
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Memo to the Commission fm Denton transmitting Draft Order and Safety Evaluation relating to Surry No. 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 26, 1980

Docket No. 50-281

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Virginia Electric and Power Company
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Sincerely,

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Order

cc: w/enclosure
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Philadelphia, Pennsylvania 19106

8005010634

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of Virginia Electric and Power Company (Surry Power Station, Unit No. 2)	))))	Docket No. 50-281

ORDER

I.

The Virginia Electric and Power Company (the licensee) is the holder of Facility Operating License No. DPR-37 which authorizes operation of the Surry Power Station, Unit No. 2 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 16512, March 19, 1979), the licensee was ordered to show cause:

(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 letters dated February 22 and March 21, 1980, the licensee requested the startup of Surry Power Station, Unit 2. This request is based on the completion of all pipe stress reanalysis and all resulting modifications installed prior to startup for all stress problems originally run on the SHOCK 2 computer program.

Technical Support for these conclusions is provided in the "Report of the Reanalysis of Safety-Related Piping Systems, Surry Power Station, Unit 2" dated February 22, 1980 and the references contained therein.

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 62 pipe stress problems which required reanalysis as a result of the March 13, 1979 Show Cause Order. Seventeen problems required hardware modifications. Of these 17 problems, seven 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 482 pipe supports inside containment. Of these supports, 165 required modifications, and about half 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 attached NRC Staff's Safety Evaluation, 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 220 supports outside containment, all have been evaluated. Of these 220 supports, 81 require modification. All modifications will be completed prior to startup.

The licensee will have completed the actions required by the Order to Show Cause dated March 13, 1979 prior to startup and this Order supercedes the March 13, 1979 Order.

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 shall be completed prior to startup.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Dated at Bethesda, Maryland
this 26th day of March, 1980.

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SAFETY EVALUATION BY THE OFFICE OF
NUCLEAR REACTOR REGULATION

FACILITY OPERATING LICENSE NO. DPR-37
VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION, UNIT NO. 2
DOCKET NO. 50-281

March 21, 1980

Introduction

On March 13, 1979; the Commission issued an Order to Show Cause to Virginia Electric and Power Company (the licensee) requiring that Surry Power Station, Unit 2 (facility) be placed in cold shutdown and the licensee show cause:

(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.

The licensee's response to the Order, dated April 2, 1979, stated that it will reanalyze all potentially affected safety systems for seismic loads using an appropriate piping analysis method. The licensee now requests that the Order be modified or rescinded such that the facility could be restarted based on the results of having analyzed all of the piping systems including nozzles and penetrations which previously used SHOCK 2, and all the corresponding piping supports. In support of this request the licensee provided information by the March 21, 1980 letter, and the letter and the attached report dated February 22, 1980 which documents the final results of all aspects of the analysis associated with the Show Cause Order. It also identifies modifications relating to the stress analysis of piping systems and pipe support evaluations. A list of correspondence which provides supplemental information is also contained in Appendix D of the report.

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 Seismic Category I systems at the facility analyzed with SHOCK 2 and has reported the results of such reanalyses. The basis of the licensee's start-up request is the confidence of system operability during the seismic events associated with the Design Basis Earthquake (DBE) and the Operating Basis Earthquake (OBE).

We have evaluated the results of the seismic reanalyses and 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.

Pressurizer Safety and Relief
Pressurizer Spray
Low Head Safety Injection
High Head Safety Injection
Containment and Recirculation Spray
Residual Heat Removal
Component Cooling Water
Service Water
Main Steam
High Pressure Steam
Feedwater
Auxiliary Feedwater
Containment Vacuum

The licensee has reanalyzed all 62 pipe stress problems originally analyzed by SHOCK 2. The licensee's request for start-up is based on completion of all pipe stress reanalysis and all resulting modifications installed prior to start-up for all stress problems originally run on the SHOCK 2 computer program, and is also based on completion of detail support analyses and resulting modifications installed for all SHOCK 2 problems.

Of the 62 SHOCK 2 problems reanalyzed, 17 required hardware modifications to bring the pipe stresses within allowables. These modifications consisted of 22 added, modified, or deleted supports. The modifications include those necessary to the flexibility analysis of the branch lines. Also, modification, addition, or deletion of 57 supports on 17 problems were necessary to reduce nozzle and penetration loads to acceptable levels. Most of these modifications are due to differences between as-built and original design, while the remaining was attributed, in part, to the incorrect use of intra-modal combinations in the original seismic analysis. Support modifications for these problems are listed in the report attached to the licensee's February 22 letter.

2. Soil Structure Interaction

Piping is analyzed in most cases utilizing amplified response spectra (ARS) that are developed using soil structure interaction techniques (SSI-ARS). The resultant stresses and loads are used to evaluate piping, supports, nozzles, and penetrations. Methods of soil structure interaction analysis which were acceptable to Surry Unit 1 are also applicable to Surry Unit 2. In accordance with the NRC letters of May 25, 1979 and November 15, 1979 to Virginia Electric and Power Company (VEPCO), the seismic inertial stresses and loads computed using the SSI-ARS have been increased by a factor of 1.5 for the DBE and 1.25 for OBE conditions.

3. 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 as applicable:

NUPIPE/Stone & Webster
NUPIPE/CDC

NUPIPE/Stone & Webster

In accordance with the letter of April 2, 1979 from V. Stello to the licensee, the licensee's Architect-Engineer, Stone and Webster (S&W) has submitted documentation on the computer code NUPIPE which is being used in the reanalysis of the Surry Unit 2.

S&W has stated that this code calculates intramodal and intermodal responses according to the provision in Regulatory Guide 1.92. A review of the code listing by the staff has confirmed this statement. The option used by the licensee specifies an intramodal combination consisting of the addition of the absolute value of the responses due to the vertical earthquake component and the root-mean-square combination of the response due to the two horizontal earthquake components. Additional documentation has also been submitted by the originators of this code (Quadrex) providing detailed information on the methods of modal combination.

The licensee has solved three NRC benchmark piping problems and its solutions show acceptable agreement with the benchmark solutions. In addition, it provided a confirmatory problem (No. 323A of Surry Unit 1 Safety Systems) to the Brookhaven National Lab for confirmatory solution. A comparison of the solutions demonstrates good agreement (within about 10%).

Based on these considerations we find the use of this code acceptable for seismic analysis by response spectrum techniques.

NUPIPE/CDC

In accordance with the letter of April 2, 1979 from V. Stello to VEPCO, Ebasco Services, Inc. has submitted documentation on the computer code NUPIPE/CDC which is being used in the reanalysis of the Surry Unit 2 plant.

This code has previously been reviewed and has been found to satisfy the requirements of Regulatory Guide 1.92. Ebasco Services Inc. has solved three NRC benchmark piping problems and its solutions were found to agree closely with the benchmark solutions. They have also provided a comfirmatory problem (2508A) which was solved by the Brookhaven National Laboratory. Comparison of the solutions show good agreement.

Based on these results we find the use of NUPIPE/CDC by Ebasco Services, Inc. acceptable for seismic analysis by response spectra techniques.

4. Reanalysis Methods and Results

The safety related piping systems at the Surry 2 nuclear plant have been reviewed to determine the method of analyses. Sixty two (62) 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. These problems have been reevaluated using acceptable methods. The reevaluation included a dynamic computer analysis using NUPIPE programs, which incorporated a lumped mass response spectra modal analysis technique.

The floor response spectra used in the reanalysis include the original amplified response spectra specified in the FSAR. In some cases, piping was reanalyzed utilizing ARS that were developed using SSI techniques. The peaks in the amplified floor response spectra were broadened by ± 15% in accordance 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 considerations 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 62 SHOCK 2 pipe stress problems have been reanalyzed and will be verified by Ebasco quality assurance, Stone and Webster Engineering Assurance and the licensee's Quality Assurance Program prior to start-up. 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.

In the reanalysis, the new total stress, at the point of maximum total stress in the pipe, and new seismic stress, at the same point, were taken from the NUPIPE computer runs with the seismic inertial stress magnified by a factor of 1.5 for the DBE condition for runs using the SSI-ARS, as required by NRC letter of May 25, 1979 to the licensee. For the OBE condition, a factor of 1.25 was used, in accordance with the NRC letter of November 15, 1979. Of the 62 problems 61 used the SSI-ARS and 1 used the original ARS. The stresses after the 1.5 and 1.25 magnification for the runs using SSI-ARS are below the allowable stresses.

To ensure that the pipe stress and pipe support reanalysis is performed as accurately as possible, field verification of as-built conditions has been performed. The field verification produced detailed piping isometric drawings and pipe support sketches for each support upon which reanalysis is based. All field-verified piping isometrics and pipe support sketches are independently verified by Surry Power Station quality control personnel.

The pipe supports were reevaluated in cases where the original support design loading was exceeded as a result of piping reanalysis. 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 702 supports (482 inside the containment, 220 outside the containment) on lines originally analyzed by SHOCK 2, and all have been evaluated, at least as far as identification of necessary modifications is concerned.

Of the 482 supports inside the containment 166 supports were identified to require modifications. Eighty-one supports outside containment are identified to require modification. During the reanalysis it was determined that 143 support modifications arose as a result of the "as-built" supports having deviated from the original design, whereas 104 support modifications can be qualified as due to inadequate, original seismic analysis incorporating algebraic summation technique.

Loads on attached equipment nozzles and penetrations were checked and verified to be either below the allowable values or were made to be below the allowable values by modification of supports. For all the problems in which the SSI-ARS are used, the seismic inertial nozzle loads have been increased by a factor of 1.5 for DBE per the NRC letter of May 25, 1979, and by a factor of 1.25 for OBE per the NRC letter of November 15, 1979. Of the 62 problems reanalyzed, hardware modifications were made to 17 problems due to nozzle overload. These modifications consisted of 57 added, modified, or deleted supports.

The pipe break criteria of the FSAR were reviewed in connection with the possible effect of changes of the high stress point resulting from the reanalyses. Only the main steam lines were included in the stress reanalysis for pipe break.

Each of the main steam lines has two terminal break locations, one at the containment penetration and the other at the main steam manifold. Each of the risers to the main steam relief valve headers has two terminal break locations, one at the main steam lines, the other at the tee into the main steam header. These terminal breakpoints are predetermined and are not changed as a result of the stress reanalysis.

Two intermediate break locations were originally determined based upon maximum primary plus secondary stresses. Upon reanalysis, two additional breakpoints on each of the steam lines were located. One of these points is located immediately upstream of the check valve and the other point is at the elbow just downstream of the check valve. All of these points will be included in the augmented inservice inspection program.

The piping systems and supports were designed to the allowable limits of ANSI B31.1 for the gross properties and to the limits of ANSI B31.7 Appendix F for local stress considerations per the FSAR criteria.

The safety related piping system supports and attached equipment, where the original analysis used an algebraic intramodal summation technique, have been reanalyzed 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.

5. Conclusion

The licensee has demonstrated that SHOCK 2 is the only method of analysis used for the facility's safety related systems which combined 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 support modifications, will be acceptable when compared with the FSAR allowables and the manufacturer's specified load criteria.

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 Surry 2, assures that safety related systems will withstand both the OBE and the DBE loading conditions. We therefore conclude that there is reasonable assurance that the facility can operate without endangering the health and safety of the public. This assurance is based on the following factors:

(1) All of the affected safety systems have been reanalyzed (piping, supports, nozzles, and penetrations) and were found either acceptable as presently designed or will be modified prior to startup.

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

Based on the above, we conclude that the conditions of the Show Cause Order of March 13, 1979, have been met.

Date: March 21, 1980.

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Date: March 21, 1980.