Docket File
50-280

AUGUST 22 1979

Docket No. 50-280

REGULATORY DOCKET FILE COPY

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. 1. The enclosed Order also confirms and requires certain commitments made by Virginia Electric and Power Company including a commitment to complete reanalysis of piping supports outside containment within 60 days of the date of plant start up.

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. 1 can safely withstand the effects of seismic events should they occur in the area. The basis for this action is set forth in the Order.

Sincerely,
Original Signed by
H. R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Order

cc: w/enclosure
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Docket File 50-280
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NRC FORM 318 (9-76) NRCM 0240 ★ U.S. GOVERNMENT PRINTING OFFICE: 1978 - 265 - 769

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Docket No. 50-280

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Office of Nuclear Reactor Regulation

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D.G.Eisenhut

OFFICE	DOR:ORB1	DOR:STSG	~~DOR:DSS~~	NRR	NRR	
SURNAME	JDNeighbors:jb	WTRussell	~~JMiller~~	EGCase	HRDenton	
DATE	08/ /79	08/ /79	~~08/ /79~~	08/ /79	08/ /79	

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Senior Vice President - Power
Virginia Electric and Power Company
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Sincerely,

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

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Order

cc w/enclosure:
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OFFICE➤	ORB#1:DOR	STSG:DOR	DSS:DOR	ONRR		
SURNAME➤	D. Neighbors	W. Russell	J. Miller	E. Case		
DATE➤	08/ /79	08/ /79	08/ /79	08/ /79		

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

August 22, 1979

Docket No. 50-280

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Senior Vice President - Power
Virginia Electric and Power Company
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Office of Nuclear Reactor Regulation

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The Honorable H. Harris
Congress of the United States
Washington, D. C. 20515

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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

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 plant startup. 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, August 1 and 21, 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 492 pipe supports inside containment. Of these supports, 51 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 22, 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 538 supports outside containment, 170 had been reanalyzed as of August 21, 1979. Of these 170 supports, 14 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., 4 of 492 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. All supports outside containment associated with high and low head safety injection, containment and recirculation spray, and auxiliary feedwater systems have been reanalyzed insuring operability of these priority systems.

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 support modifications outside containment identified in Table 4.1.B of the licensee's August 1, 1979 report as supplemented August 21, 1979, shall be completed prior to startup.

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

4. 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.

5. 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

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 22nd day of August, 1979.

SAFETY EVALUATION BY THE OFFICE OF

NUCLEAR REACTOR REGULATION

FACILITY OPERATING LICENSE NO. DPR-32

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NO. 1

DOCKET NO. 50-280

August 22, 1979

Introduction

On March 13, 1979; the Commission issued an Order to Show Cause to Virginia Electric and Power Company (licensee) requiring that Surry Power Station, Unit 1 (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 is reanalyzing all potentially affected safety systems for seismic loads using an appropriate piping analysis method. The licensee requested 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, all piping supports inside containment, and a portion of the piping supports outside containment. In support of this request the licensee provided information by letters dated March 30, April 2, 23, 24, 27, May 2, 22, 24, 30, June 4, 8, 12, 15, 19, 25, August 1 and 21, 1979. The licensee indicated that piping restraints in 19 problems needed to be modified based on its reanalysis to date.

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 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 Spray & Relief
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 Vacumm
Fire Protection
Diesel Muffler Exhaust

The licensee has reanalyzed all 63 pipe stress problems originally analyzed by SHOCK 2. In addition, the licensee reanalyzed 12 SHOCK 0 problems and 6 problems which were originally done by hand calculation. The reanalysis of those later 18 problems was not required by the March 13, 1979 Show Cause Order. All supports inside containment were reanalyzed and modifications will be completed prior to startup. A portion of the supports ouside containment have been analyzed and the remainder will be reanalyzed within sixty (60) days of the date of the Order allowing startup.

Of the 63 SHOCK 2 problems reanalyzed, 19 required hardware modification to bring the pipe stresses within allowables. These modifications consisted of 63 added, modified, or deleted supports. Also, modifications to supports on 4 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, except in Problems 508, 517, 727 and 526C where the overstress condition was attributed, in part, to the incorrect use of intramodal combinations in the original seismic analysis. Support modifications for these problems are typically as follows:

(1) Problem 743 - Low Head Safety Injection (LHSI) System. Involves a shim to close a gap between a restraint and pipe.

(2) Problem 548A - Containment and Recirculation Spray (CRS) System. Install snubber.

(3) Problem 745 - CRS System. Modify and replace restraints.

(4) Problem 746 - High Pressure Steam System. Modify and add restraints.

(5) Problem 508 - Residual Heat Removal (RHR) System. Add three vertical and one lateral restraints.

(6) Problem 540 - RHR System. Remove rod hangers. Anchor was added.

(7) Problem 744/754 - CRS System. Modify and replace restraints.

(8) Problem 562 - CRS System. Change a rod hanger to a spring hanger and add a vertical restraint in place of a spring hanger.

(9) Problem 727 - LHSI System. Install Snubber.

(10) Problem 735 - HHSI System. Install two restraints.

(11) Problems 766 - Component Cooling Water (CCW) System. Install two horizontal restraints and remove one axial restraint.

(12) Problem 481/507 - CCW System. Install two lateral restraints and two guides.

(13) Problem 480/488 - CCW System. Same as problem 481/507.

(14) Problem 509 - CCW System. Install one vertical restraint.

(15) Problem 605 - CCW System. Install four lateral restraints.

(16) Problem 526C - CCW System. Change spring hanger to vertical restraint. Install lateral restraint.

(17) Problem 2527/2529 - CCW System. Remove an anchor. Add supports on heat exchanger to make it behave as an anchor.

(18) Problem 527C - CCW System. Remove an anchor. Shim supports on heat exchanger to make it behave as an anchor. Add vertical restraint.

(19) Problem 517 - CCW System. Add vertical support. Add lateral restraints. Add a snubber. Add a vertical restraint.

Most of the problems were reviewed in detail by the NRC staff during meetings with the licensee on June 21 and 22, 1979, and July 18 and 19, 1979.

2. Soil Structure Interaction

By letter dated March 30, 1979, the licensee stated its intent to use soil structure interaction amplified response spectra (SSI-ARS) in reanalyzing the piping systems for those cases where the original amplified response spectra did not give satisfactory results. The licensee requested our review and stated that this approach was similar to that approved for Surry 3 and 4. The NRC approved in concept the use of SSI-ARS by letter dated April 13, 1979. Based upon review of the licensee's information submitted by letter dated May 24, 1979 as discussed below and our independent analyses, we informed the licensee by letter dated May 25, 1979 that SSI-ARS was acceptable.

The amplified floor response spectra (ARS) for three levels in the containment, base mat, operating floor and spring line were computed using the multi-layered elastic half space method and the finite element methods. The results of these analyses were compared for frequency and acceleration of the floor response spectra. The elastic half-space method gave acceleration values which were larger than the finite element method for the operating floor and the spring line. The finite element method gave accelerations slightly higher than the elastic half-space method for the containment base mat. Since no piping systems are located at and would not use the base mat spectra for analysis, it was concluded the elastic half-space method would be used for the reevaluation because that would be conservative. The time history used for this comparison was the original design time history used in the original design of the plant along with the original damping values.

The same floor response spectra were generated for the Regulatory Guide 1.60 requirements anchored at 0.15 g along with the Regulatory Guide 1.61 damping values for comparison with the original earthquake input requirements. The time history and the damping values are considered as a consistent set of design parameters. The comparison of the FSAR design requirements and the Reg. Guide 1.60 and 1.61 set of values show that the responses are very consistent and that the original FSAR design requirements would be adequate.

A study of the effects of the variation of the soil properties was undertaken. The response spectra for the three locations in the containment building were computed for five (5) variations of the soil properties. Variation one considered the computed strain dependent properties using the best estimate of the in situ properties as input to computer code SHAKE; variation two used the in situ properties plus 50% as input to the computer code SHAKE; variation three used the in situ properties minus 50% as input to the computer code SHAKE; variation four considered the first iteration value of the computer code SHAKE using the in situ properties as input; and variation five used the measured values (low strain) of the soil properties. This study indicated that the response of the structure to the variations in the soil properties is essentially limited to the amplitude of the floor response spectra. The peaks of the floor response spectra occur at the same place, but have different values of acceleration. For comparison purposes, two curves were constructed using the ratio of variation five to variation one response spectra, and the ratio of variation two to variation one response spectra, plotted for the operation floor and the spring line of the containment. These ratios were plotted for all values of period from 0.0 to 0.8 sec.

The ratios of variation two to variation one for the operating floor range from 1.05 at .14 sec. to 1.75 at 0.19 sec. and for the spring line 1.0 at 0.12 sec. to 1.09 at 0.29 sec. The ratios of variation five to variation one for the operating floor range from 1.1 at 0.10 sec. to 2.55 at 0.33 sec. and for the spring line from 1.2 at 0.1 sec. to 2.95 at 0.29 sec. After considering the variation of the measured in situ soil properties and accounting for uncertainty in the computer code SHAKE's prediction of strain dependent soil properties, we judged that the value of the floor response spectra acceleration would be not greater than 50% more than the floor response spectra acceleration calculated using the variation one of the soil properties. This being the case and since the licensee had already finished a large number of stress computations for the piping system using the response spectra based on the soil properties determined in variation one, it was determined that an increase of the values of the response spectra already used in piping stress calculations by a factor of 1.50 would be acceptable. This increase in the acceleration value for the floor response spectra would result in a conservative re-analysis.

To further verify that this increase (1.5) is conservative, the staff conducted an independent study of the variation of soil properties used in the dynamic analyses. First the staff confirmed the adequacy of the average soil properties selected by the licensee and then considered parametric studies of these properties. The results of this effort indicated that a variation of ±25 percent for the input shear modulus (G max) would accommodate uncertainties in the in situ soil properties. The results of this variation appear to bound the possible range in soil properties based on staff experience with other site studies.

Therefore, the licensee's studies for ±50 percent and the increase (1.5 factor) in the response spectra are conservative.

Because the soil shear moduli used in the generation of ARS depend upon the level of strain induced by earthquake motion, the ARS are not in direct proportion to the maximum ground acceleration. Therefore, an investigation of the effects of earthquakes smaller than the DBE was also undertaken.

For the purpose of this study, ARS's were computed for various average strain compatible shear moduli, each due to a peak horizontal ground acceleration ranging from 0.15 to 0.05 g.

The licensee has provided the resulting family of ARS at the operating floor which show the DBE spectrum to envelope the other spectra due to smaller earthquakes. This demonstrated that the effects of DBE are not exceeded by those of smaller earthquakes.

Therefore, based on its review of the above information the staff concludes that the stresses in piping due to the DBE are not exceeded by those due to smaller earthquakes.

The computer codes used in the re-analysis for the soil structure interaction were:

1. SHAKE
2. PLAXLY
3. REFUND
4. KINACT
5. FRIDAY

The computer code SHAKE is a public domain program and was used to compute only the strain dependent properties of the supporting soil under the structures. Because this code was only used to compute soil properties no further verification is necessary.

The computer code PLAXLY is a proprietary code and was qualified by comparison to the existing public domain computer code FLUSH. Amplified response spectra for the containment operating floor computed by both codes were compared.

The computer code REFUND computes the frequency dependent compliance functions for a multi-layered elastic half-space. This code is a proprietary code and was qualified by comparing the results of a sample problem with the results published in the literature.

The computer code KINACT is a proprietary code and is used to compute the translation and rotation time history at the base of the structure from the design time history applied at the free ground surface. This code was qualified by comparing the results of a sample problem to the results of the computer code PLAXLY.

The computer code FRIDAY uses the results of REFUND and KINACT to compute the floor response spectra for each mass point in the mathematical model of the structure. The code is a proprietary program and was qualified by comparing the results of a sample problem with the results of the public domain program STARDYNE.

The comparisons of the results for the above codes were favorable and are, therefore, acceptable by the current acceptance criteria.

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/analysis methods as applicable:

PSTRESS/SHOCK 0 (Initial 3 Versions of SHOCK 1)
Static Analysis Methods
NUPIPE

PSTRESS/SHOCK 0

This code was used for 12 safety related system problems and although it did not algebraically sum responses, the code was not equivalent to current practice. The licensee, therefore, reanalyzed these systems with the NUPIPE code.

Static Analysis

Methods used for design of the piping at the Surry Plant not subjected to computer seismic analysis were based on simple beam formulations which, in essence, controlled seismic stress levels through use of pre-established seismic spans. These simple beam formulations were utilized to calculate maximum allowable spans based upon an assumed acceleration factor of 1.5 times the peak acceleration obtained from the response spectra. In calculating the maximum span lengths, it was conservatively assumed that a longitudinal pressure stress of 4,000 psi and a maximum deadweight stress of 1,500 psi were present in the pipe. This combined value of 5,500 psi was subtracted from the allowable stress (1.8 S_h for pressure and deadweight and seismic) to obtain a seismic allowable stress.

Calculating maximum spans by this procedure results in maximum allowable spans greater than the deadweight spans recommended in ANSI B31.1. Thus deadweight governs and provides a greater number of supports resulting in closely spaced restraints. To minimize effects of concentrated weights, restraints were placed as required at valves and other concentrated masses.

For Surry Unit 1, piping 6 inches in diameter and smaller was generally analyzed using the simplified static method, with the option of utilizing more rigorous methods available to the analyst.

Piping 2 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.

As described above, the stress analysis was performed by assuming many simple supported straight beams, the spans of which are governed by dead load spacing requirements of ANSI B31.1. The piping fundamental frequencies associated with these maximum allowable spans (9.7 to 13.6 cycles per second) are not in resonance with the building in which they are located (2 to 8 cycles per second). The method of equivalent static analysis outlined in this procedure has been compared with the NRC's Standard Review Plan 3.7.2 and is found to be acceptable.

NUPIPE

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 plant.

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 responses 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) 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.

4. Reanalysis Methods and Results

The safety related piping systems at the Surry 1 nuclear plant have been reviewed to determine the method of analyses. Sixty three (63) 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 algebraic intramodal response combination technique was used in the design 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 included 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 63 SHOCK 2 pipe stress problems have been reanalyzed and verified by Stone and Webster Engineering Assurance and the licensee's Quality Assurance Program. This together with all of the 6 original hand calculations and all of the 12 SHOCK 0 problems reanalyzed completed the entire scope of piping stress reanalysis. Based on our review of the computer codes being used for reanalysis, independant 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.

The reanalysis included those SHOCK 2 problems involving the reactor coolant system boundary and the supports associated with those problems. Since the reactor coolant system boundary is inside containment and all of the supports which must be modified will be modified prior to startup, there is no potential for a loss-of-coolant accident in the event of a DBE.

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 runs using the SSI-ARS, as discussed in Section 2. Of the 63 problems 56 used the SSI-ARS and 7 used the original ARS. The stresses after the 1.5 magnification for the runs using SSI-ARS are below the allowable stresses. A pipe stress reevaluation summary provided by the licensee was reviewed by the staff to confirm that for the pipe stress problems reanalyzed, the total stress values were all below the allowable stress. Included in this summary was a listing of the original total, original seismic, new total, new seismic, and the allowable stresses for the pipe stress problems that have been reanalyzed.

At the request of the NRC, its consultant, EG&G performed audit pipe stress calculations of five Surry 1 problems using the NUPIPE computer code. The results of the EG&G audit compare favorably with the results of the licensee's results.

The piping support designs for affected system piping were 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 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 1030 supports on lines originally analyzed by SHOCK 2; of these, all 492 supports are located inside containment have been evaluated and subject to modification as identified in Table 4-2 of the licensee's August 21, 1979 submittal, are acceptable.

Fifty-one of these 492 supports were identified to require modifications. Fourteen supports outside containment are currently identified to require modification. There are approximately 400 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 having deviated from the original design. Only four piping analyses can be qualified as due to inadequate, original seismic analysis incorporating algebraic summation technique.

A NRC staff evaluation of pipe supports inside containment which required modification compared the percentage of the originally calculated support load with respect to the ultimate capacity of the supports. As of July 31, 1979 results indicate 4 supports did not have at least a factor of safety of 2 to ultimate. Ten other supports based upon preliminary calculations may not have a factor of safety to ultimate of 2. The licensee is continuing analysis of these problems, however, these 14 and the remainder of the 65 supports discussed above will be modified prior to startup.

Based on the results to date, we expect other supports may be found that will not have a minimum factor of safety of 2 to ultimate. However, if support reanalysis indicates this we will require the licensee to inform the NRC of the results of reanalysis within 24 hours and that the affected system be considered inoperable as specified in the facility Technical Specifications until the necessary modifications are implemented or a reanalysis assuming support failure is completed.

There are 112 supports outside containment which are associated with high and low head safety injection, containment and recirculation spray and auxiliary feedwater systems. The integrity of these supports assure that ECCS and systems necessary for maintaining hot shutdown will be capable of withstanding a design basis earthquake. The licensee has completed these analyses and any necessary modifications will be implemented prior to startup.

Loads on 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 reanalysis will be 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 were reviewed in connection with the possible effect of changes of the high stress point resulting from the reanalyses.

Results of the evaluation of the effect the reanalysis has on the FSAR pipe break criteria show that no new whip restraints are required. Therefore, we find that the reanalysis has not changed the pipe break protection.

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 systems supports and attached equipment, where the original analysis used an algebraic intramodal summation technique, have been, or are to be 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. <u>Conclusion</u>

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 indicated that modifications in only four problems were necessary as a result of the algebraic summation problem. These modifications are identified in Section 1. The Licensee will complete all modifications inside containment prior to start of 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 outside containment which exceed 50 percent of ultimate capacity (or twice rated load for snubbers) 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 detemined that the application of these techniques, at Surry 1, 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 outside containment which was originally seismically analyzed with the SHOCK 2 program has been reanalyzed and, subject to modification, is acceptable.

(2) All of the affected safety systems inside containment have been reanalyzed (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.

(3) The review of 487 supports inside containment identified only 4 calculated support loads exceeding 50 percent of ultimate capacity (10 other supports are still under review, however all supports are being modified). It is therefore, reasonable to expect that few supports outside containment would exceed 50 percent of ultimate capacity.

(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 analysis and will implement necessary modifications prior to startup for the supports associated with high and low head safety injection, containment and recirculation spray and auxiliary feedwater systems. These systems assure that ECCS systems and systems necessary for maintaining hot shutdown will be capable of withstanding a design basis earthquake.

(6) The licensee has committed to complete all the support reanalysis outside containment within sixty (60) days of the date of the Order.

(7) The probability of an earthquake exceeding the design basis earthquake during the sixty (60) day period 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 reanalyzed for both OBE and DBE.

(8) The NRC will require prompt notification and either resolution by reanalysis of the piping system assuming a failed support or modification of the affected support, if reanalysis of a support indicates loading in excess of 50 percent of ultimate capacity (or snubber loading greater than twice rated capacity).

Based on the above, we conclude that the licensee has shown cause why Surry 1 can be operated during completion of reanalyses required by the Show Cause Order of March 13, 1979.

Date: August 22, 1979