Docket File

AUG 8 1979

Docket No. 50-334

Mr. C. N. Dunn, Vice President
Operations Division
Duquesne Light Company
435 Sixth Avenue
Pittsburgh, Pennsylvania 15219

Dear Mr. Dunn:

The Commission today has issued the enclosed Order for termination of the proceeding initiated by the Show Cause Order dated March 13, 1979, for the Beaver Valley Power Station, Unit No. 1. The enclosed Order also confirms and requires certain commitments made by Duquesne Light Company during the course of the reanalysis.

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

Sincerely,

Roger S. Mattson for

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Order

cc: w/enclosure
See next page

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

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Roger J. Mattson for
Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Order

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of DUQUESNE LIGHT COMPANY OHIO EDISON COMPANY PENNSYLVANIA POWER COMPANY (Beaver Valley Power Station, Unit No. 1)	)))))))	Docket No. 50-334

ORDER

I.

The Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company (the licensee) are the holders of Facility Operating License No. DPR-66, which authorizes operation of the Beaver Valley Power Station, Unit No. 1 (the facility) at power levels up to 2652 megawatts thermal (rated power). The facility, which is located at the licensee's site in Beaver County, Pennsylvania, 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 is due to the fact that the technique of algebraic summation can be non-conservative. The safety implications of algebraic summation resulted from the possibility that an earthquake, of the type for which plants must be designed, could cause a reactor coolant system pipe

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rupture as well as degrade the emergency core cooling system and other systems 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:

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 March 31, 1979. In that response the licensee stated that it was reanalyzing all potentially affected safety systems for seismic loads using an appropriate method which does not sum loads algebraically.

By letter dated June 19, 1979, the licensee submitted a document entitled, "Report on the Reanalysis of Safety-Related Piping Systems for Beaver Valley No. 1 Unit," dated June 15, 1979. Revisions to this report were submitted by letters dated July 11, 18, and 27, 1979. In their letters, the licensee requested that the Commission's March 13, 1979 Order, which requires the plant to remain in a shutdown condition, be modified to permit operation of the unit for a period of six to seven weeks at which time the plant would be shut down for refueling. This request is based on the licensee's finding acceptable results of the reanalysis of the safety-related piping and supports (except as described below) for the Design Basis Earthquake (DBE) loading condition and on their commitment 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 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 0.06 g). This commitment is required only until such time that the reanalysis for the OBE loading condition, and any necessary modifications, is completed.

The exceptions to the completion of safety-related systems reanalysis involves the Fuel Pool Cooling and Purification System (FPCPS), the River Water

System (RWS) and the OBE conditions. The FPCPS is defined in the FSAR as a safety-related system. However, since the facility has not completed its first nuclear fuel cycle there is no spent fuel in the fuel pool and the FPCPS is not required to be operational. The failure of this system, in the unlikely event of an earthquake, will have no effect on the public health and safety, plant operation, or plant integrity. For the purposes of the March 13, 1979 Show Cause Order, this system is not included in the reanalysis required for facility startup. The licensee has committed to complete the reanalysis of the FPCPS using acceptable analysis techniques and complete any necessary modifications to supports before spent fuel is placed in the pool.

The River Water System has an overstress condition in two branch connections located on the discharge line in the turbine building. Failure of these branch connections in this location will not deprive any component of necessary cooling water and will not affect the functioning and structural integrity of any safety-related systems or components. The RWS at this point is downstream from the coolant supply to vital safety components and since the RWS is a once-through system, failure of this portion caused by a seismic event is clearly not a safety concern. A portion of the discharge line of the Raw Water Pumps that supplies cooling water to the turbine plant has not been reanalyzed. Although this portion of the piping was originally seismically analyzed using algebraic summation, this portion of the line does not perform a safety-related function and since it is located in the forebay of the intake structure its failure will not affect the functioning of any safety-related systems or components. The licensee has committed to complete the reanalysis of the RWS and make any necessary modifications prior to startup following the refueling outage.

The licensee has committed to reanalyze the safety-related piping to the OBE conditions and until that reanalysis is completed, to shut down and inspect the facility if a seismic event occurs which results in accelerations greater than an acceleration level of 0.01 g. Shutdown is the response required by 10 CFR Part 100 for the OBE (0.06 g). This commitment essentially resets the OBE for the plant at 1/6 its previous valve and assures that no degradation of piping, supports, or nozzles will occur which might affect their capability to withstand the DBE. The accelerometer alarm is annunciated in the control room. The staff finds the 0.01 g for shutdown and inspection to be an acceptably conservative level for resumption of operation and until the OBE reanalysis is completed.

By letter dated July 23, 1979, the licensee requested the March 13, 1979 Show Cause Order be terminated in its entirety based on the stated commitments and criteria in that letter and in Chapter Seven of the licensee's June 15, 1979 report. This was based on the fact that the licensee has completed the reanalyses for the DBE loading condition, including required modifications, of all safety-related systems with the exception of the Fuel Pool Cooling and Purification System and the River Water System. The commitments by the licensee in their July 23, 1979 letter would provide a consistent and more detailed "calculations-of-record" of all piping systems and supports where computer analyses for pipe stress are required. Although this additional effort is not intended to provide an increase in the safety of the plant and is not required by the NRC for facility startup, the effort will establish a record by which the licensee can expedite facility modification in the future.

The licensee has stated its intent to complete the calculation-of-record effort before startup following the first refueling outage.

Based on the above, the licensee concludes that the analyses and modifications completed to date and commitments made in the July 15, 1979 (as revised) Report and in the July 23, 1979 letter demonstrate that, good cause has been shown: (1) why the suspension of facility operation should not be continued and the facility be permitted to operate and (2) the March 13, 1979 Show Cause Order should be terminated in its entirety.

The licensee's analyses for operation are being performed using the SHOCK 3 and NUPIPE-SW computer codes, which combine earthquake responses in a manner acceptable to the NRC staff. The reanalyses have resulted in some stresses calculated above allowable. In such cases, the licensee has recalculated the stresses using soil structure interaction (SSI) methodology with a 20 percent increase in the seismic acceleration between the fundamental periods of 0.4 to 0.55 sec. The staff required this 20 percent increase to be applied to each pipe run after computer calculation of stress and support loads in order to ensure an added factor of conservatism. This methodology, with the 20 percent increase, was approved by the NRC staff in its letter to the licensee dated May 25, 1979.

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 the intramodal responses by the absolute value of response due to the vertical earthquake excitation to the (SRRS*) combination of the responses due to the two horizontal earthquake components. The intermodal components are calculated by the SRRS method.

NUPIPE-SW

This code combines intramodal** responses by the SRRS method and combines intermodal responses by SRSS or absolute sum for closely spaced modes. (NUPIPE-SW and SSI methodology will be the basis for the calculation-of-record effort).

The NRC staff has determined that an algebraic summation of responses was not incorporated into any of the above listed codes. The NRC staff has further concluded that these codes are acceptable for analyzing the facility piping.

Based on the NRC staff's Safety Evaluation, Attachment A, the staff finds that all safety-related piping systems including the reactor coolant system, engineered safety features, emergency core cooling systems, and all piping systems required to achieve and maintain safe shutdown or required for accident mitigation have been reanalyzed and any required modification implemented.

*SRSS - Square Root of the Sum of the Squares.
**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.

IV.

The licensee to date has completed all of the actions identified in paragraphs 1 and 2 of the Order to Show Cause dated March 13, 1979, except for (1) the Fuel Pool Cooling and Purification System, (2) the River Water System, and (3) the OBE conditions. In addition, the licensee has provided (1) justification for plant operation without the FPCPS in service and with overstressed areas in the RWS, (2) commitments for reanalyses of the FPCPS before spent fuel is placed in the fuel pool, (3) commitments to reanalyze and modify the RWS prior to startup following the refueling outage and (4) commitments to shut down the facility and inspect all affected systems if seismic accelerations at the site accelerometers exceed 0.01g.

The licensee has, pursuant to paragraph 3 of the Order, shown cause why operation of the facility should not remain suspended. In the July 23, 1979 letter, the licensee has also requested the March 13, 1979 Order be terminated.

The licensee's answer to the Order did not request a hearing. On April 2, 1979, the Pennsylvania Public Utility Commission (PUC) filed a request for a hearing and petition for leave to intevene. On April 9, 1979, the PUC amended the April 2, 1979 petition to state that it was requesting a hearing only if one or more of the following conditions exists:

1. Any other party is granted hearing.
2. It is determined by the staff of the Nuclear Regulatory Commission that an extended period of cold shutdown for Beaver Valley Unit No. 1 shall be necessary in order to make safety related modifications.
3. The Nuclear Regulatory Commission staff determination vis-a-vis the shutdown at Beaver Valley Unit No. 1 is not forthcoming within a reasonable period of time.

With respect to these Conditions:

1. No other party has requested a hearing.
2. Since this Order provides for operation, it does not mandate an extended shutdown to make modifications required as a result of this Order.
3. The NRC staff has not received any objection from the PUC regarding the reasonableness of the time within which this action was taken.

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: effective this date, the proceeding initiated by the March 13, 1979 Order is terminated and the following commitments of the licensee are confirmed and required:

1. The spent fuel storage pool shall not be used to store spent fuel until an acceptable analysis and any necessary modification have been made to the Fuel Pool Cooling and Purification System.

2. The reanalysis of and modifications to the River Water System shall be completed prior to startup following the first refueling outage.

3. The Beaver Valley Unit No. 1 shall be shut down if the site accelerometers exceed 0.01 g and the licensee will inspect all safety-related piping systems which have not been reanalyzed and shown to be acceptable at the 0.06 g level of the OBE. Prior to resuming operations the licensee will 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

Roger J. Mattson

Roger J. Mattson, Acting Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland this 8 day of August, 1979

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

DUQUESNE LIGHT COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 1

DOCKET NO. 50-334

INTRODUCTION

On March 13, 1979, the Commission issued an Order to Show Cause to the Duquesne Light Company (the licensee) requiring that Beaver Valley Power Station, Unit No. 1 (the facility) be placed in cold shutdown and the licensee 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.

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 safety and non-safety related systems at the facility.

The licensee's response to the Order, dated March 31, 1979, stated that they will reanalyze the affected facility piping systems using an appropriate piping analysis computer code which does not combine loads algebraically. Further, they stated that they will make any appropriate modifications to the affected facility piping systems which they determine to be necessary based on results of the analysis. The licensee requested that, upon completion of the reanalysis of and any necessary modifications to the affected piping systems required to assure safe shutdown and accident mitigation capability of the Engineered Safety Features and the Emergency Core Cooling System (ECCS), the facility be permitted to resume operation pending completion of reanalysis of the balance of the affected piping systems and any necessary modifications of the remaining affected piping systems. In support of this request, the licensee provided information as attachments to letters dated April 10,

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19 and 25, May 23, June 11 and 19, and July 11, 18, 23, and 27, 1979 and Stone and Webster has provided information as attachments to letters dated March 22 and 30, April 3, 6, 13 and 27, May 18, 1979, and June 4 and 18, 1979.

DISCUSSION

In this section of the Safety Evaluation, the actions conducted by the licensee and its conclusions regarding those actions are discussed. The NRC staff's Safety Evaluation of these actions and conclusions is set forth in the Evaluation section of this Safety Evaluation.

Systems

The licensee has identified 184 pipe stress problems that used SHOCK 2. Of these 184 problems, 63 were check runs of hand calculations. These 63 problems are listed in the licensee's submittal in Appendix B, Table B-2. The static analysis method, i.e., hand calculations, is discussed and evaluated later in this Safety Evaluation Report (SER). The remaining 121 pipe stress problems identified by the licensee for which SHOCK 2 was the calculation of record are in the following systems:

Reactor Coolant System
Safety Injection System
Quench Spray System
Recirculation Spray System
Charging and Volume Control System
Residual Heat Removal System
Component Cooling Water System
River Water System
Main Steam System
Main Feedwater System
Diesel Generator Exhaust
Fuel Pool Cooling and Purification System

The licensee has approached the reanalysis effort in a two phase program namely: (1) systems, or portions of systems, required for plant operation that are acceptable based on the current stress reanalysis results described in this Safety Evaluation, and (2) systems, or portions of systems, that are not currently required for operation and will be addressed in the licensee's long term effort.

Of the above listed systems that used SHOCK 2, the licensee has stated that all of these systems, with the exception of the Fuel Pool Cooling and Purification System (FPCPS), and a portion the River Water System (RWS), have been reviewed and found to be acceptable for operation.

Reanalysis Methods and Results

The piping was reanalyzed using the response spectra modal analysis technique. The piping was modeled as three dimensional lumped mass systems and included considerations of eccentric masses at valves and appropriate flexibility and stress intensification factors (SIF). The resultant stresses and loads from the reanalysis were used to evaluate piping, supports, nozzles, and penetrations. The computer codes used to perform the reanalyses were NUPIPE-SW or SHOCK 3. The acceptability of these codes is discussed later in this SER. The floor response spectra used as input in the reanalyses included the original amplified response spectra (ARS), as specified in the licensee's Final Safety Analysis Report (FSAR), and ARS developed using current soil-structure interaction (SSI) techniques. SSI methodology is discussed in greater detail later in this SER. The peaks on the original ARS and new SSI-ARS were broadened $\pm 25\%$ on the frequency to account for variations in material properties and approximations in modeling.

Reanalysis results as of July 27, 1979, show that with the addition of three supports, pipe stresses for 111 out of a total of 116 affected problems are within their allowable value of 1.8 S_h for the Design Basis Earthquake (DBE) loading case. The total stresses for 26 of these problems do not include stresses due to DBE seismic anchor movements (discussed later in this Safety Evaluation). Ninety problems indicate stresses lower than the 1.2 S_h allowable for the Operating Basis Earthquake (OBE) loading condition. Two of the problems do not include stresses due to OBE seismic anchor movements. The discharge lines of the quench spray pumps and part of the recirculation spray piping, both inside and outside containment, were seismically analyzed by NUPIPE for DBE and water hammer loads to an allowable of 2.4 S_h previous to the present reanalysis effort. These lines contain four of the five problems that show stresses above 1.8 S_h. The licensee states that they will reanalyze these using an allowable value of 1.8 S_h. The other problem with a calculated stress greater than 1.8 S_h is Problem No. 122 in the River Water System. There are two unreinforced branch connections on a segment of piping inside the turbine building that show an overstress condition.

Nozzle and penetration loads have been re-evaluated based on the results of the piping reanalysis. Of a total of 131 nozzles, 117 have been evaluated and found to be acceptable for both the OBE and DBE, and the remaining 14 are acceptable for the DBE. All 58 penetrations contained in the affected problems have been evaluated and found acceptable for the OBE and DBE.

As mentioned above, three additional pipe supports were determined to be required in order to maintain pipe stresses within allowable values. Reanalysis results also indicated that seven existing supports required modification. The problem number, system, support designation, and reason for the addition/modification are discussed below:

Problem No. 833, Reactor Coolant System - Vertical snubber added. As-built differences minor. The additional snubber was required for two reasons: First, there was significant load reduction or offset due to the algebraic summation performed in the original SHOCK 2 analysis. Second, a more conservative stress intensification factor was applied in the reanalysis.

Problem No. 217, Component Cooling Water System - Addition of one support, consisting of two snubbers. This additional support was required for the same reasons the snubber was added to problem 833. The increase in stresses when the loads were combined by the square root of the sum of the squares (SRSS) is the primary reason that this support had to be added.

Problem No. 270, Component Cooling Water System - H-56 being removed and new support being added adjacent to this location. One of the lugs to which H-56 is attached is overstressed locally due to deadweight alone. The new support will eliminate the local overstress condition by utilizing a pipe clamp. The original design for H-56 was not adequate.

Problem No. 653B, Reactor Coolant System - Supports H-30, H-31, and H-107 required modification. Stiffener plates were added to these supports and additional welding was done on the snubber bracket of H-107. These modifications are a result of changes made to the supports during plant construction, (i.e., the as-built support details were not accounted for originally).

Problem No. 653B, Reactor Coolant System - H-38 was modified to remove one direction of restraint. Originally the support was a two-directional restraint acting in both the North-South (N-S) and East-West (E-W) directions. Reanalysis showed the N-S members to be overstressed. The N-S restraint was removed and the problem rerun with acceptable results. The problem was reanalyzed with NUPIPE and modeling changes resulted in higher loads at the support. In the original SHOCK 2 analysis the degrees of freedom were restrained and, therefore, lower than actual support loads were indicated.

Problem No. 123, River Water System - H-32 angle members required stiffener plates due to increase in upward vertical load. Increased load resulted from seismic anchor movement (SAM) case which was inadvertently ommitted in the original analysis.

Problem No. 123, River Water System - H-33 required removal of lateral restraint. This support was installed as a 3-way restraint although the original piping analysis called for a 2-way restraint. The reanalysis showed that the lateral loads due to SAM overloaded the support.

Problem No. 123, River Water System - H-309 required additional stiffener plates to the structural steel. The old and new loads were approximately the same, however the original design was not adequate.

IE Bulletin 79-02 dated March 8, 1979, revised June 21, 1979, on "Pipe Support Base Plate Using Concrete Expansion Anchor Bolts" provides direction on the requalification of base plates and anchor bolts. This reanalysis effort on pipe stress and support integrity interfaces with Bulletin 79-02 at the base plate/anchor bolts. The licensee has stated that if results indicate new supports are needed or existing supports require modification, the base plates and anchor bolts shall be designed/evaluated incorporating I&E Bulletin 79-02 criteria. Additionally, field inspections will be performed on those existing base plates being modified in order to ensure bolt integrity.

Including the seven required modifications and the three additions, there are 1063 supports on lines within the scope of the reanalysis effort. Of these, 677 (including the seven modified and three added) have been evaluated and found acceptable based on FSAR criteria. Of the remaining 386, 384 are acceptable when the SAM load is removed from the DBE loading condition. The remaining two have not, as of August 1, been accepted. However, the licensee believes that there is sufficient

analytical information available for these remaining two supports to exercise engineering judgment in determining that they will be found acceptable using the same criterion applied to the 384 already found acceptable. Several snubbers have been found acceptable for a one time load equal to the DBE load.

Several of the total of 1063 supports have been evaluated for the DBE loading condition only. The licensee has stated that if a seismic event occurs which results in an acceleration greater than 0.01 g, the plant will be shut down for inspection of those supports and piping systems which have not been shown to be fully acceptable for the OBE case. The facility accelerometers and recording start at a setpoint of 0.01 g. The licensee has committed to checking the seismic instrumentation for proper operation prior to startup.

The FSAR states that break locations have been postulated for the main steam and feedwater systems inside and outside containment. The licensee states that the reanalysis results show that the highest intermediate stress points occur in those areas where the lines are fully restrained by existing pipe whip restraints and, therefore, no additional restraints are required.

Field Verification of As-Built Conditions

The licensee states that field verified piping fabricator isometric drawings provide the basis for program inputs for the pipe stress analyses. Beginning in September 1974, and completed prior to facility startup, pipe stress analysts and pipe support designers walked down all Category I (seismic) piping systems and checked for piping configuration, support location and type. The results of this effort were documented and became part of the permanent plant record. Licensee personnel verified the accuracy of a portion of these piping isometric drawings during March and April of 1979, subsequent to the Order to Show Cause.

Verification of Computer Codes and Analysis Methods

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 codes NUPIPE-SW and PSTRESS/SHOCK 3 which are being used in the reanalysis of the Beaver Valley plant.

NUPIPE-SW

S&W has stated that NUPIPE-SW 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. An option also exists for users which specifies an intramodal combination consisting of the addition of the absolute value of the responses due to the vertical earthquake component and the square root of the sum of the squares (SRSS) 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.

S&W has solved three benchmark piping problems provided by the NRC and NUPIPE-SW solutions show acceptable agreement with the benchmark solutions. In addition, S&W provided a confirmatory problem (No. 101) to the Brookhaven National Laboratory (BNL) for confirmatory solution. A comparison of the solutions demonstrates good agreement (within about 10%).

PSTRESS/SHOCK 3

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

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

Static Analysis

Much of the 6 inch and smaller Category I piping at Beaver Valley Unit 1 was analyzed using simplified static methods. The methods were intended to keep the fundamental piping frequencies out of the range of the fundamental structural frequency by establishing span lengths between supports. Calculations were based on simple beam formulations. Tabulations relating various spans, nominal pipe sizes, and acceleration levels to actual pipe stress levels were

provided for use by the analyst. The acceleration applied to the piping was dependent upon where the piping fundamental frequency was relative to the structural frequency. Calculated seismic stress was based on an assumed three component earthquake. Support loadings were based on standarized loadings enveloping the various loading conditions. Nozzle loads were calculated based on similar, simplified methods.

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

Small bore piping analyzed by a simplified static method were subjected to a NUPIPE dynamic analysis. The results demonstrated the applicability of the method and standardized support loads.

Soil Structure Interaction

The amplified floor response spectra (ARS) for three levels in the containment; i.e., 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 in the frequency range of interest. Since no piping systems would use the base mat spectra for analysis, it was agreed that the elastic half-space method would be used for reevaluation. The time history used for this comparision 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 0.125 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 variations of the soil properties. Variation one was 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 in situ properties as input; and variation five used the measured values (low strain) of the soil properties. This study indicated that variations in the soil properties causes a small variation in the frequency of the peaks and a small variation in the amplitude. The peaks of the amplified floor response spectra are broadened by ±25% on the frequency. This peak broadening would envelope the variations in frequency of each peak. The enveloping procedure also accounts for the variation in amplitude by using the maximum amplitude of the variation one and two.

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 0.125 to 0.01 g. The licensee has provided the resulting family of ARS's 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 and 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

EVALUATION

Systems

We concur with the licensee's evaluation that the FPCPS, and those portions of the RWS that have either not yet been reanalyzed or do not meet the acceptance criteria are not necessary for operation.

The FPCPS does not peform any accident mitigation function nor is it required to achieve or maintain a safe shutdown condition. The function of the FPCPS is necessary only if there is spent fuel in the fuel pool. Since Beaver Valley Unit No. 1 has not completed its first nuclear fuel cycle there is no spent fuel in the fuel pool. The function performed by the FPCPS therefore is not required for interim operation. The licensee has committed to make any modifications to this system necessary as a result of the reanalysis prior to placing any spent fuel in the fuel pool.

There are two outstanding items to be completed on the RWS during the long term effort. These involve a portion of the RWS in the Intake Structure that has not been analyzed and a portion of the RWS discharge piping in the turbine building that has an overstress condition.

The portion of the RWS that has not been reanalyzed is the portion of the Raw Water Pump discharge line (30"-WR-175-151-Q_3) that runs under the floor in the forebay of the intake structure. The Raw Water Pumps supply cooling water to the turbine plant and are not required for accident mitigation or to achieve and maintain safe shutdown. Additionally, failure of this line would not affect the operation of the River Water Pumps also located in the Intake Structure which are necessary for safe shutdown and accident mitigation. The discharge line of the RWS (30" WR-17-151-Q_3), has been found to have an overstress condition at two unreinforced branch connections. The RWS discharges to the main condenser discharge tunnel in the turbine building. The overstressed branch connections are located in the turbine building and failure of the RWS discharge line at this location would not affect the function of any safety related systems or equipment. The RWS at this point is downstream from the coolant supply to vital safety components and since the RWS is a once through system, failure of this portion caused by a seismic event is clearly not a safety concern. We concur with the licensee's determination that this overstress condition is acceptable for operation. However, the licensee has committed to modify these branch connections prior to startup following the refueling outage.

As a part of the continuing effort the licensee will reanalyze the discharge piping of the Quench Spray and Recirculation Spray Pumps and their associated spray distribution headers for the OBE loading condition. This piping was analyzed with NUPIPE for the DBE plus water hammer loads, however SHOCK 2

is the calculation of record for the OBE case. The suction piping for both the Quench Spray and Recirculation Spray System for which SHOCK 2 was the calculation of record have been reanalyzed and found acceptable for operation. The discharge piping for both spray systems is acceptable for operation and will be reanalyzed for the OBE in the long term. From a systems consideration, we find the licensee's evaluation acceptable and sufficient to permit operation.

Reanalysis Methods and Results

The three dimensional lumped mass response spectra modal analysis technique employed in the reanalysis is an acceptable method. The three components of earthquake response have been acceptably combined by the SRSS method. The analyses also considered eccentric masses at valves, (including correct weights of VELAN 6 inch check valves, as stated in the licensee's response to I&E Bulletin 79-04), appropriate flexibility and stress intensification factors, and support flexibility.

Static Analysis

In addition to the dynamic analysis (computer analysis) technique, we have also reviewed the static analysis method used for 6 inch and smaller piping. Conservative weights had previously been assumed for the VELAN 3 inch check valves. The methods of equivalent static analyses employed are similar to the procedure described in Section 3.7.2 of the Standard Review Plan and are acceptable.

Results of the pipe stress reanalysis show that, after the addition of three supports and the modification of seven others, stresses in all but five piping problems are below the allowable for the DBE loading case. In accordance with the FSAR, the allowable is taken from the 1967 version of the ANSI B31.1.0 Code including addenda up to and including June 30, 1971. Additionally, DBE seismic anchor movement effects have been neglected for some piping problems and many supports. Consideration of only the inertial portion of the DBE load, i.e., neglecting DBE seismic anchor movement effects, is in accordance with Section III of the ASME Code, to which nuclear power plant piping is designed today, consistent with current practice and, therefore, acceptable. The licensee has committed to 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 inspect those piping systems and supports which have not been shown to be fully acceptable for the OBE case (ground acceleration of 0.06g). This commitment essentially resets the OBE for the plant at 1/6 its previous value and assures that no degradation of piping, supports or nozzles will occur which might affect their capability to withstand the DBE. The staff finds the 0.1 g for shutdown and inspection to be an acceptably conservative level for resumption of operation and until the OBE reanalysis is completed. The accelerometer alarm is annunciated in the

control room. Therefore, we find the evaluation of the facility capability to withstand a DBE acceptable for resumption of operation.

Four of the five problems with stresses exceeding 1.8 S_h are on the Quench and Recirculation Spray Systems. However, all stresses are below 2.4 S_h, the currently accepted stress allowable for this loading condition for new plants. Additionally, the licensee has committed to reanalyze these problems using an allowable of 1.8 S_h. Since the stresses on the problems are currently based on the old ARS, we believe that reanalysis with the new SSI-ARS will result in stresses below 1.8 S_h.

The fifth problem showing stresses above 1.8 S_h is Problem No. 122 in the River Water System. The two overstressed branch connections are within the turbine building and we found this condition acceptable from a systems consideration previously in this evaluation. All stresses in the remainder of this piping are below their 1.8 S_h allowable value.

At the request of the NRC, INEL/EG&G* performed audit pipe stress calculations on five Beaver Valley problems using the NUPIPE-II computer code. The results indicate all pipe stresses to be within allowable values. A direct comparison between the SHOCK 3/NUPIPE-SW stresses calculated by the licensee's consultant and the EG&G audit results was not made. Further, the results of the audit calculations indicate that seismic stresses may be significantly altered depending on support stiffnesses used and which method of seismic response combination (algebraic vs. SRSS) is employed. If piping natural frequencies are close to the natural frequencies of the building, relatively small (e.g., 10-15%) shifts in piping frequencies can result in significant increases in accelerations. These frequency shifts may occur when support stiffness is varied. The problems analyzed with NUPIPE-SW incorporated realistic support stiffness values (e.g., 10^5 - 10^7 lb/in) and, therefore, the calculated frequencies are approximately correct. For those problems analyzed with SHOCK 3, a tabulation of 10 and 15% frequency shifts and corresponding accelerations indicate that, when the SSI-ARS is considered, the current pipe stress reanalysis results are reasonable or conservative.

The licensee has identified three pipe stress problems whose results are based on the original ARS and have natural frequencies in an area of the new SSI-ARS that is not enveloped by the old ARS. However, results of a detailed examination of the current stress level to allowable value

*Idaho Nuclear Engineering Lab/EG&E (consultant to the NRC).

indicate sufficient margin available if the accelerations increased to those corresponding to the new SSI-ARS.

Based on the above evaluation, we find the piping stresses resulting from the reanalyses acceptable.

Results of the re-evaluation of all 131 nozzles and 58 penetrations are acceptable.

The support evaluation indicates all but two are acceptable, following the modifications required on seven. The licensee believes that, based on engineering judgment, both will be acceptable upon further evaluation using the DBE inertia load only and neglecting the DBE seismic anchor movement load. This criterion is acceptable to the staff as previously stated in this evaluation. The licensee has committed that, prior to resumption of operation, these two supports will be determined acceptable or they will be modified to make them acceptable. We believe this commitment adequately addresses the acceptability of these two supports.

Some hydraulic snubbers have been found acceptable for a one time load corresponding to the DBE load. The basis for their acceptability is an April 11, 1979 letter from R. J. Masterson of ITT Grinnell Corporation, manufacturers of the snubbers, to M. Pedell of S&W. Prior to Cycle 2 operation, the licensee will have to quantify the loading and corresponding acceleration level that the snubber could be subjected to and remain within FSAR acceptance criteria and revise the facility technical specifications to reflect this condition. If this load or the acceleration level is exceeded, the snubber will be tested for operability prior to continuing operation or returning to power. With this commitment, therefore, we find this criterion acceptable. The licensee may replace snubbers qualified by this criterion to make them fully conform to FSAR criteria.

Other than the two supports and the snubbers discussed above, for the DBE case all remaining supports are in accordance with original design criteria, AISC Code and WRC Bulletin 107 for local stresses, and are acceptable. In addition, the licensee has also committed to make any modifications to supports, excluding hydraulic snubbers themselves, discussed above, required to meet FSAR acceptance criteria for both the OBE and DBE loading cases. Also, prior to return to power for the start of Cycle 2 operation, the licensee has committed, by letter dated July 23, 1979, to complete the seismic reanalysis of all safety related piping using the NUPIPE-SW computer code and the new SSI-ARS. All piping stresses, support loads, and nozzle and penetration loadings will be evaluated for both the OBE and DBE load conditions, based on their respective acceptance criteria. All acceptance criteria will be in accordance with the FSAR or exceptions acceptable to the NRC staff, discussed above. The use of the NUPIPE-SW computer code and the SSI-ARS has been found acceptable by the NRC, as

evaluated later. Further, the use of this computer code and these response spectra curves will adequately address the potential problems due to support flexibility and the new SSI-ARS not being enveloped completely by the old ARS.

Results of the evaluation of the affect 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 of the facility.

Field Verification of As-Built Conditions

An April 18-21, 1979, inspection of Beaver Valley 1 by NRC inspectors from the Office of I&E resulted in no items of noncompliance being identified within the scope of the inspection. The inspection results are discussed in a May 25, 1979, letter from R. Carlson of I&E to C. Dunn of DLC. The inspectors examined for accuracy the as-built safety related pipe supports and pipe system drawings. Based on the information on the subject provided by the licensee, as discussed previously, and on the results of the I&E inspection, we believe that the reanalyses accurately reflect the as-built condition of the plant.

Verification of Computer Codes

As discussed previously, the staff's review of the NUPIPE-SW and PSTRESS/SHOCK 3 computer code listings confirm that the codes calculate intramodal and intermodal responses as stated by S&W. Also, solutions to the benchmark and confirmatory problems demonstrate good agreement with the benchmark and BNL confirmatory solutions. Based on these considerations we find the use of these codes acceptable for seismic anaysis by response spectrum techniques.

Soil Structure Interaction

The soil-structure interaction (SSI) analysis for the Beaver Valley Power Station, Unit No. 1, has been reviewed against the current staff positions. As discussed previously in this SER, the staff required studies 1) comparing ARS generated using the FSAR time history and damping values and Regulatory Guide 1.60 and 1.61 requirements, 2) of the effects of varying the soil properties, and 3) investigating the effects of earthquakes smaller than the DBE. Based on the results of these studies, we conclude that the method used to develop the new SSI-ARS is acceptable.

The computer codes used to develop the SSI-ARS were SHAKE, PLAXLY, REFUND, KINACT, and 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. 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. KINACT is a proprietary code and is used to compute 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.

To verify that the licensee's proposed ±25% peak broadening of the amplified response spectra was conservative, the staff conducted an independent study of the variations in soil properties which were used in the dynamic analyses. First the staff checked the validity of the average soil properties selected by the licensee and confirmed that the values were appropriate. The staff then conducted a parametric study using the computer code SHAKE with variations of ±50% from the best estimates of in situ soil properties. The results of this study indicated that a variation of +50% for the input shear modulus would cover the uncertainties in the in situ soil properties. The lower -50% variation in properties was not considered representative of the soils at the plant site. It was also determined that the establishment of the actual lower variation bound was not necessary because the amplified response spectra of the best estimate properties and the +50% variation were shown to essentially envelope the spectra curve of the -50% variation in the frequency range important in pipe stress analysis.

Based on staff studies and a review of the licensee's work, the staff concluded that the proposed ±25% peak broadening was reasonably conservative with one exception. Design ground motions in the free-field at foundations level were previously established by the applicant by calculating the site response due to a number of earthquakes, then enveloping the calculated site response with an assumed site independent response. This procedure resulted in design motions with frequency dependent conservatisms, with minimum conservatisms occurring at the natural frequency of the soil deposit overlying the rock. In an effort to add conservatism in the natural period range of the foundation soils, the staff required at least a 50% increase in spectral acceleration above the response curve which was developed using the best estimate soil properties. The natural periods of the foundation

soils was estimated to range from 0.4 sec. to 0.55 sec. The staff's requirement essentially caused a 20% increase in the amplified response spectra above the peak broadened spectra in the natural period range of the foundation soils.

Based on the above, and since the SSI-ARS used took into account the staff's recommendation to increase the spectral accelerations by 20% in the period range of 0.4 to 0.55 sec., we find acceptable the ±25 peak broadening.

CONCLUSION

Based on the above discussion and evaluation, we conclude that Beaver Valley Power Station, Unit No. 1, may resume power operation. This conclusion is based on the required modifications to seven supports, the addition of three others, and the two supports not yet found acceptable are determined acceptable or modifications to make them acceptable being completed prior to startup.

Date: [illegible] 8 1975