RESPONSES TO QUESTIONS

IN

JUNE 26, 1979 LETTER

FROM

SENATOR GARY HART

7911270208

QUESTION 1: When performing cost/benefit analyses of alternatives in NEPA reviews, how does NRC factor into those analyses costs such as those entailed in shutdowns (whether voluntary or by order or license conditions) of reactors because of safety concerns?

ANSWER:

The cost associated with unscheduled shutdowns, whether voluntary or by order or license conditions, is factored into NRC's cost/benefit analyses through the forced outage rates. For generic purposes planned outage rates (POR) of 12% to 15% and forced outage rates (FOR), including shutdown to remedy safety concerns, of 9% to 11% are representative for nuclear units. POR of 10% to 12% and FOR of 10% to 14% are representative of large coal units with sulfur removal equipment. These are equivalent to about a 75% to 80% availability factor for nuclear and about 76% to 81% availability factor for coal units. Because of distribution system reliability and other considerations, capacity factors are generally a few percentage points less than the availability factors. Thus a capacity factor of about 60% is reasonable for comparing the economics of coal and nuclear. This is consistent with the historical capacity factor for large base loaded coal and nuclear plants.

Generally, the unit costs of electricity generation for nuclear and coal in NRC's NEPA reviews are calculated for a range of capacity factors. Figure 1-1 shows the sensitivity of generation cost as a function of capacity factor for both coal and nuclear in the New England and North Central (MT, ND, SD, WY, CO and UT) regions. These two regions represent the extremes for the contiguous United States. The capacity factor at which the cost of generation is equal for coal and nuclear is about 60% in the North Central region and 40% in the New England region.

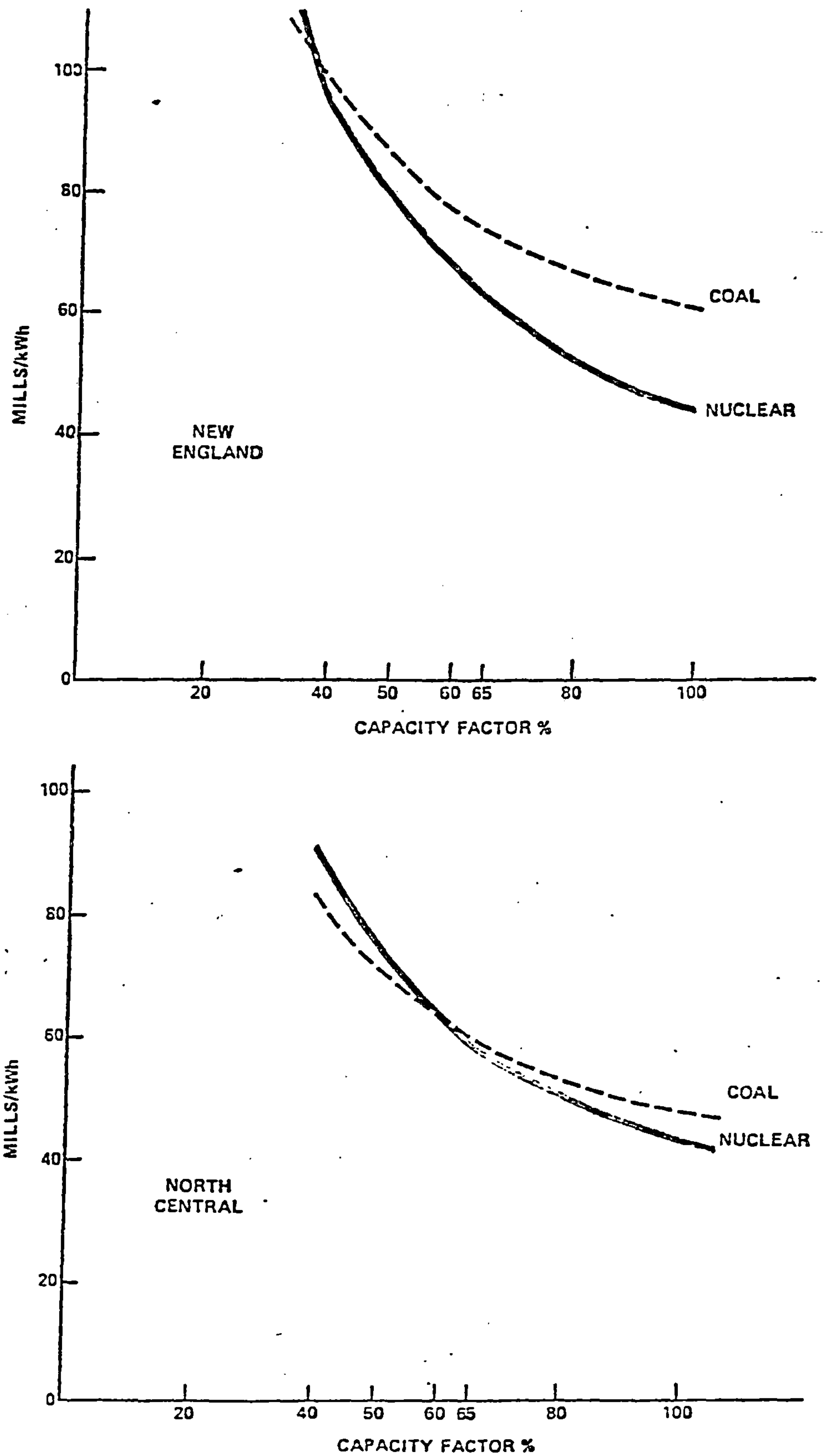


FIGURE 1-1

The Effects of Capacity Factor on Unit Cost (Initial Year of Operation - 1990 Mills/kWh)

QUESTION 2: How has NRC assured that the codes being used in the reanalysis of seismic design produce valid results?

ANSWER:

The Office of Nuclear Reactor Regulation has instituted a code verification and confirmatory analysis program whereby the licensees and/or their contractors were required to solve a set of piping benchmark problems devised by the NRC staff. These problems consist of representative piping structures of varying complexity subjected to seismic loading, for which solutions were determined independently by an NRC consultant, the Brookhaven National Laboratory. The licensee-generated solutions have been compared with the benchmark solutions and acceptable agreement has been found between them.

In addition to the benchmark problems, the licensees also provided to the NRC a representative piping problem from each of the affected plants, together with their corresponding solutions. These problems were in turn solved independently by the NRC consultant, who confirmed (by comparison of the solutions) that the licensees' results were correct. This constituted the confirmatory analysis portion of the program.

As a preliminary step to the analysis program described above, the NRC staff has also reviewed the FORTRAN code listings of portions of the codes used for reanalysis and has confirmed that the analytical algorithms as programmed in these codes conform to presently acceptable methods of seismic analysis of piping structures.

These three steps (i.e. licensee verification analysis, independent confirmatory analysis, and code listing review) provide reasonable assurance that the codes used for reanalysis provide valid results.

QUESTION 3: What steps have been taken to assure other computer codes currently being used for reactor designs do not contain errors?

ANSWER:

The code verification and confirmatory analysis program described in the response to Question 2 is being extended and applied to codes used for seismic analysis of piping structures by other licensees and their contractors. In addition, a previously instituted research program at the BNL for generating benchmark problems and solutions is also being extended to obtain benchmarks for a broad variety of codes, by both analytical and experimental techniques. The use of benchmark problems and solutions for code verification is described in item (b) below.

Although computer codes used in the analysis of structures and systems other than piping are not specifically reviewed by the staff, the applicability and validity of these computer programs have been demonstrated by one of the following criteria or procedures.

(a) The computer program is a recognized program in the public domain, and has had sufficient history of use to justify its applicability and validity without further demonstration. The dated program version that will be used, the software or operating system, and the computer hardware configuration must be specified to be accepted by virtue of its history of use.

(b) The computer program's solutions to a series of test problems, with accepted results, have been demonstrated to be substantially identical to those obtained by a similar, independently written program in the public domain. The test problems should be demonstrated to be similar to or within the range of applicability for the problems analyzed by the computer program to justify acceptance of the program.

(c) The program's solutions to a series of test problems are substantially identical to those obtained by hand calculations or from accepted experimental test or analytical results published in a technical literature. The test problems should be demonstrated to be similar to the problems analyzed to justify acceptable of the program.

QUESTION 4: Please list each reactor which has been found since March 13, including the five reactors which were the subject of the hearing, to have had an error in the seismic analyses of plant design. In your response, please include:

(a) whether the reactor was shutdown because of the error;
(b) whether the shutdown was voluntary or by order;
(c) the systems involved;
(d) whether the systems are safety related or non-safety related; and
(e) the resulting corrective measure if any.

ANSWER:

At the time of the original safety review of the plants in question, specific NRC (then AEC) guidance on acceptable methods for combining seismic forces did not exist. Nuclear industry practice to combine seismic forces for piping systems varied; some design organizations used algebraic summation, others used square root sum of the squares (SRSS) and others used absolute summation methods. It has thus developed that a number of plants were designed using analysis techniques, which were accepted practice to a portion of the nuclear industry at the time (i.e., were state of the art) and are clearly unacceptable today. In December 1974, when Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components In Seismic Response Analysis", was issued providing specific guidance on acceptable methods, the staff did not review earlier plants to determine if unacceptable methods had been employed.

Our efforts to reevaluate the seismic analyses and design of piping systems have been directed at only safety related systems since these are the systems which are of importance to assure the protection of the public health and safety. The list of the plants which have been found thus far to have used the algebraic summation technique for the combination of codirectional responses to multiple earthquake input components is contained in the accompanying table, including whether or not the reactor was shut down, whether the shut down was voluntary or by order, a general description of the system involved, and any corrective measures.

PLANT	SHUTDOWN REQUIRED		EXTENT OF SYSTEMS ANALYZED USING ALGEBRAIC SUMMATION TECHNIQUE	CORRECTIVE MEASURES
	ORDER	OTHER		
Beaver Valley 1	Yes		Extensive	Complete and Order terminated 8/8/79
Brunswick 1,2	No	Voluntary	Extensive	Complete
Cook 1, 2	No	No	Main Reactor Coolant Loop and some lines inside containment	Complete
Cooper	No	No	SRV lines	Complete
Fitzpatrick	Yes		Extensive	Complete and Order terminated 8/14/79.
Ginna	No	No	Main Steam and RHR lines	Complete
Indian Point 2	No	No	10 Lines	Completed by licensee. Staff SER in preparation.
Indian Point 3	No	No	Extensive	Shutdown for refueling. All work to be completed by licensee & staff SER written prior to start up. Estimate start up - mid December.
Maine Yankee	Yes		19 lines (Initially thought to be extensive)	Complete and Order terminated 5/24/79
Millstone 1	No	No	2 systems (Control Rod Drive Exhaust and CU2 Bypass)	Complete
Millstone 2	No	No	6 systems (Volume Control Tank Changing Bypass, Nitrogen Addition, Charging, Diesel Generator Exhaust, RCP Top Root Valve Instrument, SI and Containment Spray Test Line)	Complete

TABLE FOR RESPONSE 4 (page 2 of 3) (as of 10/5/79)

PLANT	SHUTDOWN REQUIRED		EXTENT OF SYSTEMS ANALYZED USING ALGEBRAIC SUMMATION TECHNIQUE	CORRECTIVE MEASURES
	ORDER	OTHER		
Nine Mile Pt.1	No	No	7 systems (Reactor Recirculation, Shutdown Cooling, Emergency Condenser Returns, Reactor Cleanup, Reactor Drain, Reactor Feedwater CRD).	Complete
Pilgrim 1	No	Tech Spec*	Recirculation and Main Steam lines	Complete
Pt.Beach 1,2	No	No	2 CCW and 2 SW lines in radwaste system	Complete
Robinson 2	No	No	Main Reactor Coolant Loop	Complete
Salem 1	No	Immediate Action Letter	Extensive	Reanalyses and implementation of modifications in progress.
Surry 1,2	Yes		Extensive	Order permitting operation of Surry 1 issued 8/22/79. Surry 2 shutdown for steam generator repair.
Turkey Pt. 3,4	No	No	Main Reactor Coolant Loop	Complete
Zion 1, 2	No	No	Main Reactor Coolant Loop	Complete

* During the algebraic sum review, the licensee identified "as built" problems with a number of snubbers. Tech Specs required plant shutdown under these conditions.

TABLE FOR RESPONSE 4 (page 3 of 3) (as of 10/5/79)

PLANT (Under Construction)	EXTENT OF SYSTEMS ANALYZED USING ALGEBRAIC SUMMATION TECHNIQUES	CORRECTIVE MEASURES
Salem 2	Extensive (Reactor Coolant System excluded)	Reanalyses and implementation of any required modifications prior to criticality.
Forked River	Containment Spray	Reanalyses and implementation of any required modifications prior to receipt of operating license.
WNP 1, 4	ASME Code Class 1 Reactor Coolant System Branch Lines	Reanalyses and implementation of any required modifications prior to receipt of operating license.

QUESTION 5: (a) What technical standards/methods are being used to determine the adequacy of design seismic events - those existing at the time the 5 plants were licensed or those in existance at the time? If the former, please describe:

(b) The differences,

(c) The rationale for not applying modern standards, and

(d) A brief assessment of the relation between the existing seismic designs for the 5 plants and the existing standards.

ANSWER:

The analytical methods used in the reassessment of the three soil supported plants (Surry 1 & 2 and Beaver Valley) were the same standards used to assess plants applying for a license today, but the seismic inputs and other acceptance criteria were those approved in the Final Safety Analysis Reports (see attached Table). The response of the structure (buildings) to an earthquake in the original analytical method was overly conservative, therefore current and more realistic techniques were used to model soil-structure interactions. The seismic inputs, which included the original design earthquake and the associated damping values for structures and piping systems, were analyzed and compared to an analysis which used current design earthquake and the corresponding damping values. This comparison showed that the response of the structure and its equipment were essentially the same. The original damping value used for the piping is less than that required today resulting in a higher seismic load on the piping. Therefore, the original design earthquake together with the originally assigned damping values for structures and piping systems is acceptable when compared to that which is required today. Based on the assessment of the three soil-supported plant design earthquakes and their damping values, the original design earthquakes and damping values were determined to be adequate and conservative in comparision with those which would be used today.

The other two plants (Fitzpatrick and Maine Yankee) are founded on bedrock. The reanalysis of these plants was limited to a reanalysis of the piping systems and did not include a reevaluation of building structures response to earthquakes. However, in both cases the NRC staff reviewed the adequacy of the original design earthquake and structural damping values and determined that the seismic input to the piping reanalysis was acceptable. The results of the Systematic Evaluation Program's seismic review will be used as a basis for further seismic analysis at Maine Yankee.

TABLE FOR RESPONSE 5

TECHNICAL DATA

UNIT:	CURRENT DESIGN PRACTICE	SURRY 1 & 2 ORGINAL DESIGN	MAINE YANKEE ORIGINAL DESIGN	FITZPATRICK ORIGINAL DESIGN	BEAVER VALLEY 1 ORIGINAL DESIGN
EARTHQUAKE: OBE DBE VERTICAL COMPONENTS	REGULATORY GUIDE 1.60	.07 g .15 g 2/3 HORIZ. 2	.05 g .10 g 2/3 HORIZ. 2	.08 g .15 g 2/3 HORIZ. 2	.06 g .125 g 2/3 HORIZ. 2
DAMPING: STRUCTURES OBE DBE PIPING OBE DBE	REGULATORY GUIDE 1.61	 5 % 10 % 0.5 % 1.0 % *	 2 % 5 % 1.0 % * 2.0 % *	 CONCRETE / * / ** 2 % / 2% / 1% 5 % / 3% / 1% 0.5 % 1.0 %	 CONCRETE / * 2 % / 5 % 2 % / 7 % 0.5 % 1.0 %
COMPUTER PROGRAMS USED FOR PIPE DIAMETER:	IN GENERAL ALL SAFETY RELATED SEISMIC CATEGORY I & ASME SECTION III PIPING ARE ANALYZED USING COMPUTER CODES	ALL > 6" SOME < 6"	ALL > 6" SOME < 6"	ALL > 6"	ALL > 6"
		* Verification done with 0.5 % (Ref.: Seismic Design Review Report)	* 0.5/1.0 for welded steel low-stress piping between rigid supports	* Steel frame bolted/riveted ** welded	* Total soil-containment structure system

QUESTION 6: (a) How do the perceived risks associated with the error in the seismic design of the 5 plants compare with those associated with the Babcock and Wilcox plants during the first five weeks following the accident at Three Mile Island?

(b) What factors led to the shutdown of all of the former within a few days of learning of the shortcomings, while some Babcock and Wilcox plants never were shutdown?

ANSWER:

At the time the decision was made to require immediate shutdown of five plants for seismic reasons, the perceived risk was as high or higher than in the case of the other B&W plants after the Three Mile Island accident, some of which were allowed to continue operation while modifications were made.

The error in seismic design appeared significant in that a single event (of a seismic nature) could damage the integrity of the reactor coolant system thereby causing a LOCA and also preclude operation of the ECCS which is designed to protect against the LOCA. An accident and the disabling of required protective systems could occur as a result of not meeting a fundamental design criteria. On the basis of early recalculations by Stone and Webster for the Beaver Valley facility, it appeared that the problem was widespread. In the judgment of the Director of Nuclear Reactor Regulation, the problem was significant enough to recommend shutting down the affected units.

On the other hand, a single event, including a seismic one, was not known to endanger the safe operation of a B&W plant after Three Mile Island. It is true that a B&W plant did experience a real problem while a seismic event is hypothetical, but a TMI type of event was the result of several actions occurring in a particular sequence. A repeat or a similar event was judged unlikely in the very short term.

The confidence that there was no undue risk in the short term (few weeks) from B&W reactors while additional modifications were made included:

1. The high state of readiness and training of operators to cope with feedwater transients as a result of bulletins which were issued shortly after the TMI accident.

2. The lowered likelihood of relief or safety valves lifting on feedwater transients because of the reduced scram pressure setting and higher power operated relief valve setting recommended by B&W and required by the NRC.

3. The low likelihood of failure of initiation of auxiliary feedwater.

4. Evaluations performed by B&W which were stated to show prediction of the TMI voiding sequence and good cooling for several analyzed transients with failure of feedwater where high pressure safety injection systems would need to be relied on.

(Note: Although Commissioner Bradford agreed, at the time, that these and other specific modifications were prudent and provided a considerably enhanced level of assurnace, he reserved final judgment until the completion of the then-ongoing generic review of feedwater transients in B&W reactor and plant systems. Following completion of that review, staff recommendations resulted in the temporary shutdown of all other B&W nuclear power plants for additional modifications.)

QUESTION 7a: What are the recurrence frequency and magnitude of the design basis and operating basis earthquakes at each of the 5 plants?

ANSWER:

The Design Basis Earthquake or the Safe Shutdown Earthquake and the Operating Basis Earthquake (OBE) are defined in detail in Appendix A to 10 CFR Part 100. The design requirements for the OBE are such that the plant structures, systems and components necessary for continued operation, without undue risk to the health and safety of the public, are designed to remain functional. In the event of the occurrence of an earthquake, up to and including the OBE level, no regulatory action would be required. If the OBE level were to be exceeded, NRC Regulations require plant shutdown. Prior to resuming operation following this shutdown the licensee would be required to demonstrate to the Commission that no functional damage has occurred to those plant features necessary for continued operation without undue risk to the health and safety of the public. For earthquakes up to and including the SSE, it is required that the structures, systems and components necessary to assure integrity of the reactor coolant pressure boundary, the capability to shutdown the reactor and maintain a safe shutdown condition and the capability to prevent or mitigate accidents leading to unacceptable offsite exposure all be designed to remain functional. Therefore, the SSE, not the OBE, is the important earthquake level on which to focus attention from the standpoint of safety in the evaluation of the capability of a plant to withstand a seismic event and safe shutdown.

In his testimony of March 27, 1979, Mr. Denton presented some estimates of recurrence frequency of the Design and Operating Basis Earthquakes at the five plants. He indicated that the Design Basis (Safe Shutdown) Earthquake had a chance of being exceeded at each of the four sites that was of the order of 10^{-3} to 10^{-4} per year. He also indicated that the chance of the Operating Basis Earthquake being exceeded was roughly estimated to be on the order of five times that of the Design Basis Earthquake.

These numbers were based upon previous estimates of earthquake ground motion exceeding a given peak acceleration at various locations throughout the eastern United States. Because of the lower design acceleration and higher local seismicity, the Maine Yankee site appeared to be at the higher end of the risk of exceedance range. In these estimates no attempt was made to expand upon the applicability of or the uncertainty associated with these values. In response to your question we will supplement Mr. Denton's original testimony with a discussion of these factors and provide an updated and more site specific estimate of recurrence frequencies.

It should be pointed out first that probabilistic estimates of earthquake hazard (recurruence frequency) were not used in defining the original earthquake resistant design at the five plants. Those numbers presented previously by Mr. Denton and in this response represent _a posteriori_ estimates of exceeding original design ground motion parameters which were chosen in a deterministic manner.

While probabilistic estimates of seismic hazard can be made, insight and great care must be exercised in utilizing these estimates in the decision making process. Our experience indicates that absolute estimates of these hazards for a site can vary by more than an order of magnitude, depending upon the choice of input parameters and assumptions. The choice of parameters and assumptions will vary among expert seismologists. A thorough estimate of seismic hazard should systematically include these varying opinions and should account for the related uncertainty. This type of estimate could require a lengthy research program which is at or possibly beyond the state-of-the-art.

In order to answer your question at this time we can only examine those readily available studies that have included the different site regions in their estimates of seismic hazard and, by interpolation and extrapolation, provide gross ranges of return periods (recurrence intervals) for the different design and operating basis earthquakes (see Table 7-1). These studies include those performed by individual members of the U. S. Geological Survey, the Canadian Department of Energy, Mines and Resources, the Applied Technology Council and other seismologists.

The most important data base upon which all of these estimates ultimately rest is the historic (non-instrumental) record of the felt effects of earthquakes. Converting these felt effects (earthquake intensity) into instrumentally determined earthquake magnitudes or ground accelerations that may be useful in design is itself a complex and often controversial task. This is in great part due to the shortage of appropriately measured earthquake motion in the eastern United States. The magnitude scale utilized in the table below is that developed by Professor Otto Nuttli of St. Louis University and is roughly equivalent to the Richter Magnitude (developed for California earthquakes) in the magnitude range of interest.

The return periods listed below are for earthquake ground motions used in the design of the five plants. They do not represent return periods for exceeding structural design limits or for failure of any plant component. The ground motion at each site is specified in terms of two parameters:

1) Peak Ground Acceleration (PGA) - the most common description of earthquake ground motion. This is the parameter used in most of the studies and can, therefore, be determined directly. In our March 27, 1979 submittal to this committee our initial judgement of earthquake recurrence was based solely upon chances of exceeding the peak ground acceleration.

2) Response Spectrum (RS) - a method of characterizing the variation of level of ground motion as a function of frequency. It can be shown that at very short periods (high frequency) the value of the response spectrum is about the same as the peak ground acceleration. The spectra used for design are usually standardized shapes developed from studies of actual earthquakes. Over the years, as the number of earthquakes recorded has increased, these standardized shapes have changed. When the 5 plants were originally designed, the response spectrum shape used at that time was different than that which our present regulatory guides specify. In relating earthquake response spectra to either intensity, magnitude or peak acceleration, we have assumed that the level of ground motion indicated in the current regulatory guide spectrum is appropriate. As a result the estimated return periods associated with the response spectra shown on Table 7-1 differ from those relying solely upon peak acceleration.

TABLE 7-1

Estimated Magnitudes and Return Periods[1] for Design Earthquakes Corresponding to the FSAR Peak Ground Acceleration (PGA) and FSAR Response Spectra (RS) for the Four Reactor Sites
(From Extrapolations and Interpolations of Readily Available Studies)

	DESIGN BASIS EARTHQUAKE (SSE)		OPERATING BASIS EARTHQUAKE	
Plant Site	Magnitude	Return Period (yrs)	Magnitude	Return Period (yrs)
Surry				
PGA[2]	roughly 5.3	greater than 2500 yrs	roughly 4.8	roughly 500 to 2500
RS	roughly 4.8	roughly 500 to 2500	roughly 4.4	roughly 80 to 600
Beaver Valley*				
PGA	roughly 5.2	roughly 1000 to 10,000	roughly 4.8	roughly 250 to 2500
RS	roughly 5.0	roughly 800 to 7000	roughly 4.6	roughly 150 to 1500

*NOTE - the Beaver Valley site has soil conditions which, when properly accounted for, could amplify the ground surface accelerations resulting from bedrock motion. This could lower the magnitude of the design earthquake and lead therefore to shortened return periods.

Fitzpatrick				
PGA	roughly 5.3	greater than 1000	roughly 4.9	roughly 400 to 6000
RS	roughly 4.8	roughly 300 to 4000	roughly 4.3	roughly 100 to 800
Maine Yankee				
PGA	roughly 5.0	roughly 250 to 3500	roughly 4.5	roughly 50 to 700
RS	roughly 4.5	roughly 50 to 700	roughly 4.0	roughly 20 to 100

1. Return periods are estimated average recurrence intervals for earthquakes of given size or greater over extremely long time intervals (many times the length of the return period). In no way are they meant to predict the actual occurrence of an earthquake in a given year but rather the average chance of its happening. Similarly the limits of these return periods are not meant to denote strict boundaries in which all present or future estimates will be contained. They are simply the broad band of return periods determined from the interpolation and extropolation of those studies examined. It should be emphasized that these return periods refer to earthquake occurrence alone and do not refer to assumed response of the piping or structures. They provide lower bounds for the return periods for response spectra which could cause calculated stresses in the piping to approach these values calculated in the original seismic analyses. Consideration of parameters such as the increase in allowable dampings for structures and piping, and soil/structure interaction (where applicable) would tend to provide an increase in this return period relative to the pipe stress calculated for design. Extensive analyses would be required to provide an accurate estimate of this increase for all of these plants. However, for Beaver Valley and Surry, based upon the comparison of the soil/structure interaction reanalyses using current and the licensed spectra and dampings, the return periods for earthquakes with

spectra for which calculated stresses in the piping would approach their values calculated in their reanalyses would tend to be roughly in the range of those given for the peak ground acceleration, rather than those given for the response spectra.

2. PGA refers to peak ground acceleration and RS refers to response spectrum.

QUESTION 7b: Based on the reanalyses using acceptable procedures, what are the recurrence frequency and magnitude of the earthquake that would have resulted in stresses above the allowable limit prior to any plant modifications.

ANSWER:

The review effort for the safety related piping systems on the 5 plants was focused at determining the adequacy of the systems to resist the specified earthquake design criteria and to implement any required modifications. It was not directed at determining the earthquake level at which the systems as built would reach their allowable stress limits prior to modification.

The determination that the stresses in a piping system are within allowable limits for the specified design criteria requires not only an evaluation of the stresses within the pipe itself, but also support and nozzle loads and their resulting stresses. Additionally, the seismic load is considered in combination with other loads which also produce stresses in piping supports and nozzles. This further complicates the estimation of the earthquake level at which the allowable stresses would have been reached in the unmodified condition. In addition, the seismic analyses of piping systems do not predict the exact stress levels in the piping under seismic levels. They merely provide stress magnitudes for design purposes. It is impossible to uniquely characterize the nature of the ground motion at a site as a function of earthquake magnitude and to predict exactly the seismic responses of piping systems.

Using SSE design parameters and acceptance criteria (spectra, damping, allowable stress limits, etc.), the earthquake peak ground acceleration level at which allowable stress limits would have been reached in the as-built piping systems may be estimated from the information we have to date for Beaver Valley Unit 1. For Surry Units 1 and 2, Fitzpatrick and Maine Yankee, we do not possess sufficient information regarding the stress levels predicted in the unmodified piping systems to make such an estimate.

For Beaver Valley Unit 1, given the new response spectra based on soil/structure interaction considerations, reanalysis results to date indicate that six pipe supports require modification, three snubbers must be added and at least one branch connection reinforced in order to bring all pipe stresses, support, and nozzle loads within their respective SSE allowable limits. However, many other supports could not be found acceptable until the SSE seismic anchor movement load, originally included, was removed in accordance with current ASME Code criteria. Several snubbers also could not meet original design criteria and have been found acceptable after reevaluation of their capacity. Without knowing specific magnitudes of overstress or overload conditions, and given the acceptability of removing the SSE seismic anchor movement load and the one time snubber loadings, about 95% of all calculated stresses, support and nozzle loads would remain within their allowable SSE criteria for a ground acceleration of 0.125g. Utilizing the extrapolations and interpolations discussed in the response to 7a, this would roughly correspond to an earthquake of magnitude 5.0 and would have a recurrence interval on the order of thousands of years. The other 5% would require a substantial reduction, to possibly as low as 0.05g, before they could meet their allowable limits. Utilizing the extrapolations and interpolations discussed in the response to 7a, this would roughly correspond to an earthquake of magnitude 4.6 and would have a recurrence interval on the order of hundreds of years or more. The same caveats discussed in 7a would also apply to these rough estimates.

QUESTION 8: What are the estimated costs of the shutdowns of the 5 plants in terms of dollars and barrels of oil? The underlying assumptions should be stated.

ANSWER:

NRC did not estimate a cost for the shutdown of Surry II, because it was already shut down at the time the error in seismic analysis was found.

For the other plants, the operating utilities were contacted to determine cost impacts. These costs agree with NRC calculations when the same assumptions are used.

Beaver Valley

The replacement power for the 852 MWe unit is supplied by burning coal. Assuming a capacity factor of 74%, the monthly costs are $5.1 million for coal (11.23 mills/kwh), $0.8 million for purchase of power (1.76 mills/kwh) and $0.5 million (1.1 mills/kwh) for increased cost of non-fuel operation and maintenance. A savings of nuclear fuel cost is about $1.7 million per month (3.8 mills per kwh) leaving a net cost* of $4.7 million per month or $160,000 per day.

Maine Yankee

Oil is burned for replacement power at $16 per barrel (27 mills per kwh) compared to the nuclear cost of 3.3 mills per kwh. At a net capacity rating of 830 megawatts and a monthly capacity factor of 95%, 28,000 barrels of oil per day would be required and the net cost* of replacement power would be about $450,000 per day.

Surry 1

The replacement power is supplied by burning oil at $18 per barrel (30 mills/kwh) at a net capacity rating of 822 megawatts and a monthly capacity factor of 75%, 23,000 barrels of oil per day would be required and the net cost* of replacement power would be about $340,000 per day.

Fitzpatrick

The replacement power is provided by burning oil at about $16 per barrel (27 mills/kwh). At a net capacity rating of 821 megawatts and at a 75% capacity factor, the net cost* of replacement power is about $330,000 per day and requires about 24,000 barrels of oil per day.

*The net cost is the cost of oil or coal minus the cost of nuclear fuel not consumed.

QUESTION 9: In the March 16 hearing, Mr. Denton remarked that much credit for bringing the computer error to his attention goes to the diligence of an NRC inspector who pursued the discrepancy in the results of the old and new codes. Please provide the particulars in a chronology of the surfacing of the discrepancy and an assesment of the reasons for any delays.

ANSWER:

As stated previously by Mr. Denton and reaffirmed herein, the NRC Inspector deserves much credit for actively pursuing with Duquesne Light Company and Stone and Webster Engineering the problem in pipe stress analysis. As the enclosed chronology indicates, there was persistent NRC staff effort to obtain information that would accurately define the safety issues so that appropriate actions could be taken. An assessment of the potential safety significance of the problem was considered throughout the fact finding process. The staff moved in a manner consistent with the safety significance perceived at the time based on the information provided to the NRC. When the cause of the discrepancy in the results of stress analyses was identified to the NRC staff, prompt action was taken that led to the issuance of the Show Cause Orders.

Attachment: Chronology Table

10/26/78 — Prompt report LER 78-053/01P to NRC Region I via telecon from Duquesne Light Company. Reported information received from Stone and Webster that hand calculation errors resulted in stress levels above ANSIB 31.1, 1967 but only in one case of six flow paths.

10/27/78 — Daily Report by Region I to I&E headquarters included as a reportable occurrence - inadequate piping supports during review of safety injection pipe stress analysis by the A/E (S&W), several points on the 6-inch and smaller piping were found to be inadequately supported. In the event of safety injection system operation during a DBE, 5 points could exceed the code allowable stress. A design change for safety injection piping supports will be accomplished prior to unit startup in mid-November.

10/27/78 — Written interim LER submitted by Duquesne Light Company. DLC characterized the errors reported by Stone and Webster as resulting from a hand calculation method of analysis.

10/[illegible]-11/3/78 — IE Inspection 50-334/78-30 - Region I followup on 24 hour report. Inspector raised a number of questions including: What assurance can be given to show that the calculational error applies only to the six points in question? To only the Safety Injection system? To only the Beaver Valley facility?

11/9/78 — Second interim LER submitted by Duquesne Light Company indicates that the original report was erroneous. The line stresses were thought to have been hand calculated only, when in fact they were subsequently computer calculated and found acceptable. DLC also indicated that a full report on the situation was in preparation by Stone and Webster.

11/14-17/78	IE Inspection 50-334/78-33 - Region I inspectors followup but no information available onsite.
11/16/78	Region I Daily Report indicated a rereview by A/E found that the previously reported condition was erroneous and that no inadequately supported piping existed, a full report of the situation is being prepared by the A/E and a followup to the LER will be submitted by the Licensee to NRC.
11/30/78	Followup calls to site by the IE inspector attempting to seek additional information.
12/01/78	Followup calls to site by the IE inspector attempting to seek additional information.
12/04/78	Followup calls to site by the IE inspector attempting to seek additional information.
12/05/78	Followup calls to site by the IE inspector attempting to seek additional information.
12/06/78	LER 78-53/01T-0 was submitted to NRC by licensee. Conclusion was that "corrective action has been reviewed, approved and satisfactorily completed". The report based on information supplied by Stone and Webster attributes the pipe overstress to differences between stresses analyzed by PSTRESS code and those done by the chart method. It mentions differences between PSTRESS and NUPIPE codes in force summation but does not elaborate on them. It concludes that PSTRESS used methods acceptable for Beaver Valley Unit 1 generation plants. It states that Reg. Guide 1.92 issued in December 1974 established for facilities docketed after April 1975 more conservative techniques for intramodal combinations of generalized loadings. The report states that analysis showed that only one safety injection system pipe required modification - the addition of one snubber and the redesign of one support. The attachment to this LER provided additional historical information as follows:

Duquesne Light Company reported in an attachment to the December 6, 1978 LER 78-53/01T-0 that to generate data needed for installation of a net positive suction head modification to the Beaver Valley Unit 1 safety injection system, they (Stone and Webster) decided to "code in" the six inch SI lines into a currently used computer program (NUPIPE). DLC indicated original design used the PSTRESS code. No results of an analysis at this stage were reported by DLC to NRC.

Subsequent to the above activity the attachment states the Beaver Valley Power Station was notified by a vendor that check valves in SI system were actually heavier than used in design at construction stage. This increased weight was used as input to the above NUPIPE model and found not to "affect" the piping design. The Architect Engineer (Stone and Webster) also concluded that the hanger designs need not be changed as a result of using the correct (heavier) weight for these valves. However errors were said to have been discovered in the hand calculation method. It was determined that piping analysis showed local overstress at several anchors but no overstress in "the pipe" alone.

Per attachment to LER 78-53/01T-0, a more thorough evaluation was initiated to determine if "any other annulus piping" originally designed by the chart (hand calculation) method was overstressed.

Per attachment to LER 78-53/01T-0, licensee found that SI lines had been "as-built" reviewed in 1974 and that two of the six lines had been (at that time) coded into PSTRESS (not just hand calculation method). The PSTRESS code was re-run using the correct valve weights and resulted in acceptable pipe stresses.

Also per attachment to LER-78-53/01T-0, licensee states "The models run in PSTRESS and NUPIPE are geometrically similar; however, the mass distribution and support stiffness are different. Further, the method of force summation (intra-modal) is different. NUPIPE utilizes more conservative techniques for intra-modal combinations of generalized loadings.

These newer techniques arose following establishment of Beaver Valley Unit No. 1 design criteria. In December, 1974, the USNRC published Regulatory Guide 1.92, applicable to facilities docketed after April, 1975, which required the use of the more conservative combinations. The PSTRESS methods used were accepted dynamic analysis techniques for Beaver Valley Unit 1 generation plants, and is the basis for all computerized Category I pipe stress analysis performed".

(It is NRC understanding that results were unsatisfactory on two of three lines, but snubber and support modifications on one line reduced the overstress on the second line such that no modifications on that line were necessary.)

The pre December 6, 1978 review of annulus seismic piping was limited to lines that had been previously analyzed using the hand calculation method (2-1/2 inch to 6 inch lines). 103 lines were identified, 55 were reviewed and found acceptable. Licensee noted that PSTRESS results were still available for 48 of the 103 lines from the 1974 as built review and were "acceptable".

Licensee notes its Engineering Department is "continuing a review of the architect-engineer findings".

12/11/78 Followup calls to site by the IE inspector to seek additional information.

Region I IE inspector telephoned NRR Licensing Project Manager to obtain a contact for informal discussion of technical questions.

12/12/78 Region I Daily Report - Further review of in-containment SI system piping supports identified one line requiring support modification, attributed to an error in original design calculations.

12/14/78 Regional inspector was telephoned by NRR individual who was designated as contact. Preliminary technical discussion was held about potential problems.

12/18-20/78	IE Inspection 50-334/78-34 - Region I followup on 12/6 LER. During this inspection, the inspector reviewed the detailed report submitted to the licensee by A/E and discussed the results of that review with representatives of the licensee and A/E.
12/22/78	Region I inspector discussed with NRR individuals via telephone questions he had as a result of discussions he had with S&W on 12/18-20/78. The NRC individuals involved determined that there was a possible problem.
1/18/79	Region I mailed to IE Headquarters a memorandum requesting that information be forwarded to NRR for review. The memo defined concerns to include: 1. Reconciliation of the differing analysis results to assure that the design methods used are neither incorrect nor unconservative. 2. The need for further licensee review of piping potentially affected by any incorrect or nonconservative calculation.
1/23/79	The IE Inspector provided copy of the 01/18/79 memorandum to Licensing Project Manager.
About 2/2/79	Discussion between IE inspector and NRR project manager determined that a formal transfer of lead responsibility between I&E and NRR had not been made of the 01/18/79 memorandum to NRR.
2/2/79	A formal request for DOR's Engineering Branch support (TAC form) was prepared by the project manager.
2/5/79	IE inspector was informed by IE:HQ that telephone discussion had established that NRR was working on the problem and that a formal transfer of lead to NRR would be made.
3/1/79	During a conference call to DLC and S&W, a computer run was requested for DOR review. Since S&W corporate policy was not to provide such proprietary data, a meeting was set up for S&W to bring in a computer run for DOR review at Bethesda.

3/8/79 A technical meeting was held between DLC, S&W, and the NRC staff to discuss and review the PIPESTRESS and NUPIPE codes. The NRC approached the review with the belief that the two codes were acceptable and that some modeling or input problem created the results in question. It was revealed that the PIPESTRESS code used an algebraic summation of seismic loads which in the absence of a detailed time history analysis, gave unconservative results in the seismic stresses. Management was immediately informed and a management level meeting arranged with DLC and S&W.

3/8/79 A management level meeting was held with DLC and S&W to arrange for immediate review of the Beaver Valley pipe stress analyses. Commitments were requested of S&W to identify the systems and plants involved, the inadequacies expected and the reanalysis to confirm safe operation. No definitive information was available at that time. DLC was requested to have its plant safety committee review the situation.

3/9/79 Numerous staff meetings were held at Bethesda to scope the problem with respect to the effects if a seismic event were to occur. Telecons were made to S&W on the schedule of commitments for further information on Beaver Valley. The other utilities identified by S&W as having plants with the same problem were notified. These plants and utilities were: Fitzpatrick, Power Authority of the State of New York; Maine Yankee, Maine Yankee Atomic Power Company; Surry 1 and 2, Virginia Electric and Power Company.

The Chairman was advised. Three staff members were sent to Boston to provide immediate review and analysis of results. DLC sent eight people to Boston to assist in expediting the review.

3/9/79 In view of the problems and with the Offsite Safety Review Committee concurrence, the Beaver Valley Unit 1 was placed in hot standby for the weekend by DLC to await further analyses from S&W.

3/10/79 Staff meetings continued as pieces of information were fed back from Boston. The I&E Duty Officers were advised of actions. The NSSS vendors for the plants were contacted to assure no other codes for pipe stress during that period used the same algebraic approach. A DOR Assistant Director was sent to Boston to provide management review and coordination. S&W's computer was dedicated full time to these stress calculations and extended work hours for data reduction was instituted for S&W staff. NRC options were explored and draft materials developed to support appropriate action based on the technical results becoming available on Beaver Valley.

3/11/79 Early S&W reanalysis results on Beaver Valley runs indicated problems with pipes as well (originally thought only supports). Licensees'-top management was contacted to assure action underway by all plants to identify inadequacies and obtain reanalyses of stresses in all affected safety systems.

3/12/79 Additional information from DOR staff in Boston confirmed pipe stresses above allowable and unacceptable.

Arrangements were made to brief the Commission on this matter. All the licensees were notified of a pending decision.

3/13/79 In view of the safety significance of this matter as discussed above, the Director of the Office of Nuclear Reactor Regulation proposed to the Commission that the public health and safety requires that the present suspension of operation of the facility should be continued: (1) until such time as the piping systems for all safety systems have been reanalyzed for earthquake events to demonstrate conformance with General Design Criterion No. 2 using a piping analysis computer code which does not contain the error discussed above, and (2) if such reanalysis indicates that there are components which deviate from applicable ASME Code requirements, until such deviations are rectified. The Commission concurred in the NRR Director's decision.

Prior to the NRC final decision to order the plants shutdown, the Beaver Valley Offsite Safety Review Committee recommended the facility be placed in cold shutdown based on the data and analysis recieved from S&W. The DLC ordered the plant shutdown.

3/14/79 The licensees confirmed by telecon that the Orders were received and provided times when each facility would be in cold shutdown. All facilities will be at or below 200°F by 10:40 p.m. on March 15, 1979 in conformance with the Order.

Subsequently all affected licensees were notified by telephone that the Orders were executed and that a copy would be transmitted by facsimile.

3/16-17/79 Meetings were held with Stone and Webster with the Utilities to discuss acceptable methods of analysis for interim and long term fixes of the piping and supports.

QUESTION 10: Please provide available information on the recent earthquake that occurred in the vicinity of the Maine Yankee plant. How does it compare with the operating basis earthquake?

ANSWER:

A small earthquake (magnitude 4.0) occurred on April 17, 1979 at 9:34 p.m., local time, near Brunswick, Maine and about 10 kilometers west of the Maine Yankee Plant site. The earthquake was felt over a broad area of New England and was recorded at many of the NRC-supported seismograph stations of the Northeastern U.S. Seismic Network. An intensity investigation conducted by MIT suggested that the highest intensities were MM (Modified Mercalli) V. No damaged resulted anywhere from the earthquake. At Maine Yankee the licensee informed the NRC that the earthquake was felt in the control room but not in the containment. According to the licensee, there was one operating strong motion recorder at the site (trigger set at 0.01g in the vertical direction) and the earthquake did not trigger this device.

About 30 aftershocks were recorded in the first 24 hours following the earthquake. The largest of these aftershocks was magnitude 2.8. A plot of the epicenter locations describes a cluster of events centered approximately 10 kilometers west-northwest of the Maine Yankee site. There are no known structures in the vicinity of the earthquakes which the NRC staff believes to be localizers of seismicity.

Approximately one day after the magnitude 4.0 earthquake, personnel from the Massachusetts Institute of Technology and the Lamont Observatory installed networks of portable seismographs in the epicentral area to record aftershocks. They monitored for several days and recorded only one aftershock.

Two additional aftershocks occurred on May 11 and 13, 1979. Their magnitudes were measured at 2.3 and 2.7, respectively.

The NRC provided instrumentation for a sensitive portable seismograph network in the epicentral area to attempt to detect and accurately locate any additional aftershock activity. Weston Observatory of Boston College installed these stations about June 1, 1979. The Maine Geological Survey maintains these stations and performs preliminary analysis of the records, and Weston Observatory performs detailed evaluations of the data. Very small earthquakes (about magnitude 1) were detected on June 6, 1979 (two events) and June 18, 1979 in the vicinity of the magnitude 4.0 earthquake. This portable network will remain in operation through July, 1979 and will operate after July only if there is additional activity.

The NRC staff concludes that the Maine Yankee site did not experience ground motion exceeding the Operating Basis Earthquake (OBE) from the April 17, 1979 earthquake because:

1) The maximum intensities observed at the site are associated with ground motions less than those associated with the OBE.

2) The earthquake did not trigger the strong motion recorder at the site.

As noted in the answer to question 7a, the response spectra for the OBE using current NRC regulatory guide spectra approximates a magnitude 4.0 earthquake. Given the wide range of expected acceleration levels, another earthquake of the same size as the April 17, 1979 event located near the site might equal or exceed the OBE.

QUESTION 11: One of the plants ordered shut down is the Surry Plant which served as the model PWR for the Reactor Safety Study (RSS). The RSS included an extensive design adequacy study.

(A) What was the finding of the study team with respect to the seismic design of Surry?

(B) What are the ramifications with respect to future quantitative risk assessments?

ANSWER TO 11A:

The Design Adequacy task of the Reactor Safety Study is reported in Appendix X to WASH-1400. The method of combining modal inertial forces is discussed in Section A6.3.3.1 of Appendix X (pp. X-47 - X-49). This discussion states that "The method of combining modal inertial forces in the principal directions to determine seismic stresses is not correct." However, it was the understanding of the Reactor Safety Study Group that the absolute value of the model forces were combined, rather than the algebraic values. Thus, the Reactor Safety Study concluded that the method used "... leads to conservative results...". We now know that this understanding was incorrect. The general findings of the Reactor Safety Study regarding the seismic design of the PWR analyzed (Surry) are presented on p. X-3 of Appendix X and are quoted below:

> "... 30 PWR items were examined with regard to seismic design. Of these, 25 were found to be adequate (83%). Design adequacy was not demonstrated for five items (17%) (reactor coolant pump nozzles, low head safety injection system instrumentation, recirculation spray pump outside containment, the diesel generator day tank, and the AC and DC switchgear), because sufficient information was not available to permit an assessment of adequacy to be made. For three items (the containment crane, the low head safety injection pumps, and the reactor protection system), it was found that the design was adequate in that failure is not expected under seismic excitation. However, the margin to failure was found to be less than that normally expected considering applicable code and qualification requirements because either: (1) errors were found in assumptions used in calculating stresses; or (2) seismic qualification tests were not sufficiently comprehensive or were not performed."

(Note: No seismic modifications to the Surry Power Station were made as a result of the RSS conclusions.)

ANSWER TO 11B:

The impact of seismic design deficiencies recently identified has been estimated to increase the risk and overall core melt probability by a factor of 3 to 4 over that estimated in the Reactor Safety Study. With respect to future quantitative risk assessments, this deficiency, plus analyses performed by others on seismic risk potential suggest the following:

1. A comprehensive design adequacy review is necessary when considering the response to loadings not included in the available data base, e.g., severe seismic events.

2. A definitive need exists for improved modeling of the seismic contribution to risk. In this regard, NRC has a large ongoing seismic research program which is intended to provide the information needed to define the seismic risk contribution more precisely.

QUESTION 12: Please list all nuclear power plants that have been exported from the United States that were designed with the aid of the erroneous computer code involved in the five plant shutdowns.

ANSWER:

We are unaware of any nuclear power plants exported from the United States that were designed with the aid of the computer code involved in the five plant shutdowns. However, we are aware of a number of foreign organizations which have entered into Royalty Agreements with A. D. Little, Inc. of Cambridge, Massachusetts for the right to use ADLPIPE, a computer code which utilizes the algebraic summation technique. A list of these organizations is included in the attachment to this response.

Arthur D. Little, Inc. ACORN PARK · [illegible] · TELEX 921436

April 19, 1979

Mr. Vincent S. Noonan, Chief
Engineering Branch
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Noonan: 98705

I am enclosing a memorandum which confirms the information furnished at a meeting with you and other members of the NRC Staff Monday afternoon, April 16, 1979.

I am sending a copy of this letter (and its attachments) to John G. Davis, Acting Director, Office of Inspection and Enforcement, under cover of transmittal, a copy of which is attached for your information.

A copy of this letter and its attachments are being sent to the organizations listed in Appendix II, who are ADLPIPE licensees.

As discussed at our April 16 meeting, we will verify the five bench mark problem solutions (after receipt of the problems from NRC) published in BNL-NUREG 21241-RS and BNL-NUREG-23645 utilizing the present version of ADLPIPE, February 1977, Version 3C.

If you desire any further information, do not hesitate to call.

Very truly yours,

I W Dingwell

I. W. Dingwell

sp

Enclosures Memorandum
Letter to John G. Davis from I. W. Dingwell of 4/19/79

MEMORANDUM April 19, 1979

A Brief History of ADLPIPE (see table 1)

Arthur D. Little, Inc., first prepared a program in 1952 to compute the flexibility and thermal deformations of piping systems for a private firm. An ASME paper was delivered in April 1956, "The 6X6 Matrix Method of Piping System Stress Analysis". Later during the liquid oxygen fueled ballistic missile program, Arthur D. Little, Inc., adapted this program to make dynamic analyses of missile fueling systems.

A new program, ADLPIPE, was developed in the period 1967-1968, first for the static (deadweight, thermal, external force, applied displacement) analysis of elastic piping systems. The program was written in FORTRAN and designed to be independent of the particular computer system used. The second development--also in 1968--(modification one) was for the dynamic (modal) analysis of lumped mass piping systems. The transient loading was described as a response spectra.

Following a prototype development period, a version was delivered in August 1970 which enabled the user to implement ANSI B31.7 "Nuclear Power Piping". This version could not produce a full stress report but gave stresses for particular loadings. In 1972 a version was released which enabled the user to produce a partial stress report to meet the requirements of ASME Section III. In 1972 this version was released to Control Data Corporation Cybernet. In 1973 the computation of fatigue usage factors was completed. In 1974 a version was released for the utilization of ASME Section III, Class 2. In 1975 a force time history analysis was included for the calculation of hydraulic transients. At the same time a one-dimensional thermal transient analysis was developed for the requirements of ASME Class 1.

In 1976 the automatic computation of seismic analyses in accordance with Regulatory Guide 1.92 was developed and checked. The complete matrix analysis portion of the program was rewritten based generally on the techniques of SAP IV with some improvements in the matrix storage methods. In addition, a post-processor was developed which allowed the user to make load set combinations for use in applications other than Regulatory Guide 1.92. This version was released in February 1977 and upgraded in December 1977 and September 1978.

In the period 1968 to 1973, ADLPIPE was the only computer program (which was available to the public) for computing piping response to various static and transient loads. Other programs were in use, but to our knowledge, these were proprietary and not available for general use.

From its inception, ADLPIPE could be utilized for a variety of stress calculations not involving nuclear power piping. In 1975 applications were extended to meet the requirements of chemical plant and refinery piping and petroleum transportation piping.

TABLE I

DEVELOPMENT OF ADLPIPE

1967	Development of static load version
1968	Delivery of static version Delivery of prototype dynamic version
1969	----------------
1970	Delivery of static dynamic B31.7 version
1971	----------------
1972	Inclusion of ASME Section III Class 1 Inclusion of closely spaced modes
1973	Inclusion of ASME Section III Class 1 usage factors Inclusion of Metric units
1974	Inclusion of ASME Section III Class 2 and 3, B31.1
1975	Revised input organization
1976	Force time history analysis Transient thermal analysis (one-dimensional)
1977	Inclusion of 1.92 modal summation (group method) Inclusion of post-processor for new 1.92 summation Revised matrix storage and solution
1978	----------------

Modal Analysis by ADLPIPE During the Period 1968 Through 1976

In this period of time, ADLPIPE was licensed to several clients and released beginning in 1972 to several nationwide computer service bureaus. A listing of ADLPIPE versions and documentation is given in Appendix I. The names of ADLPIPE licensees and the effective dates of the license agreements are given in Appendix II.

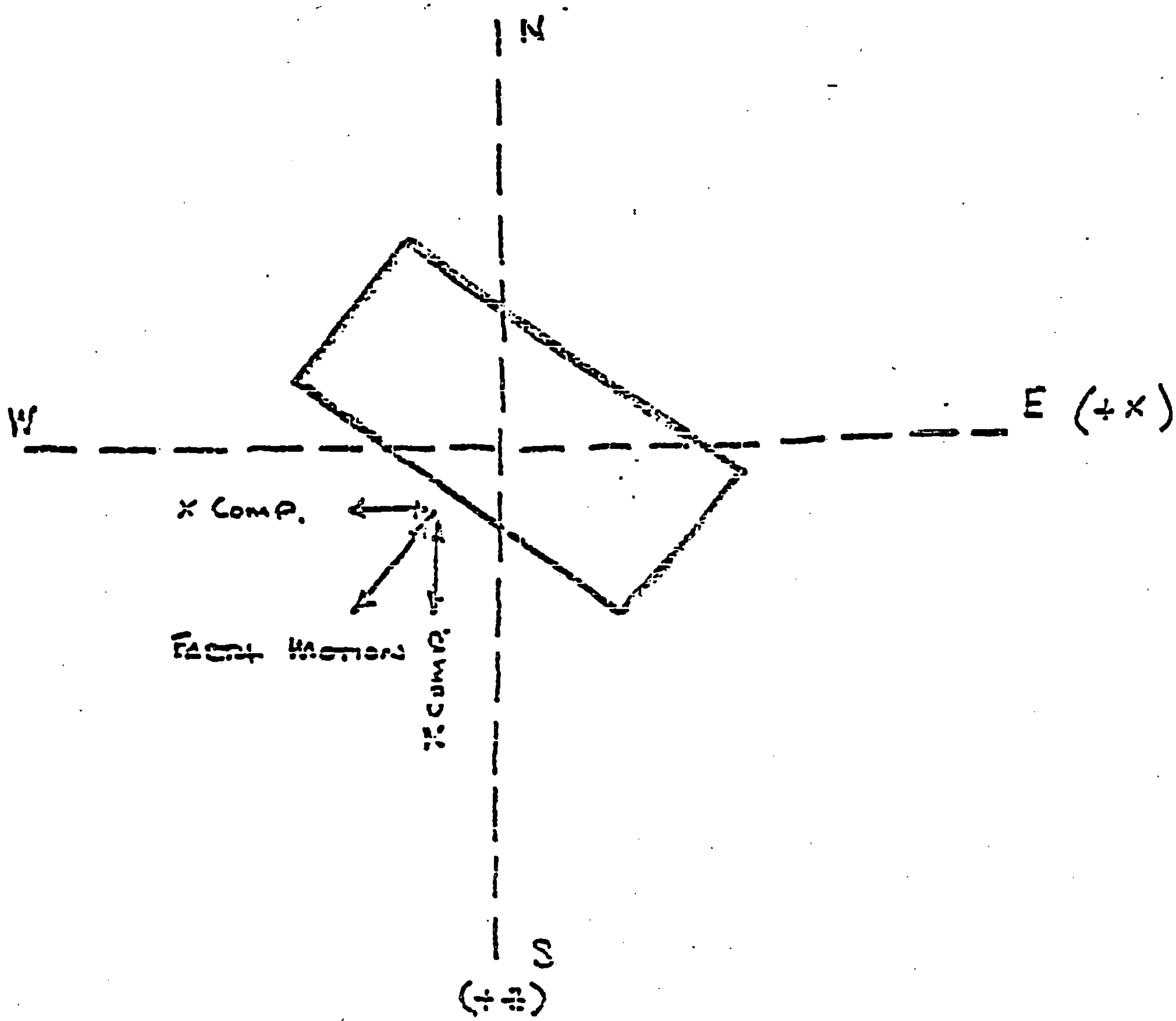
The development of the seismic analysis method was guided by available literature and the design requirements of our clients. A method of analysis was developed which was explained by two documents published in 1969. These are enclosed as Appendix III, "Modification One--Response to Ground Shock Spectra" and Appendix IV, "Development of Modal Participation Matrix for General Three-Dimension Shock Input to Lumped Dynamic System". In Appendix III on page III-5, I state "the modal amplitude, q_n, is thus evaluated as a scalar summation of the products of the nth vector of the modal participation matrix and the spectra amplitude $(D_\ell)_n$". The "spectra amplitude" means the spectra displacement components in the principal coordinates of the piping system. From these modal amplitudes, a set of displacements for each mode of response is computed. At each point in the piping system, three modal moment components are then computed, one of each principal axis. Each component was then squared and then the square root of the sum of squares was taken to combine the effect of all modes. This concept was used to model earth motion along a vector which was not necessarily aligned with a principal axis but was skew and was decomposed to three components. The reason for this development is shown in Figure 1 (page 4) where a structure is not aligned with a global coordinate system. An earthquake is assumed to act perpendicular to one wall of the structure. Mathematically, the skew axis of the earthquake is decomposed into two horizontal components in the global axes.

A user could calculate earthquake response with a vertical component and a single horizontal component if the two axes were decoupled combining several such analyses to create a worst case effect. A user could make three or more different analyses, one for each principal axis again combining the results.

Users who made a single analysis using a tri-directional earthquake would have printed out a single set of modal moments. If one isolated each response spectra component by a separate analysis and computed three sets of individual moment components, the resultant from the single tri-directional analysis would be the algebraic sum of each individual component for each earthquake directional component. The upper level would be the absolute sum of the intra-modal components. The lower level could be zero within a mode. However, it is my view that the inter-modal summation using the square root sum of the squares would not be zero and, in fact, would not vary greatly (±33 percent) from a square root sum of squares (SRSS) intra-modal summation. A numerical example is given in Appendix

FIGURE 1

UNIDIRECTIONAL EARTHQUAKE WITH SKEW COMPONENTS



V, "Dynamic Analysis by ADLPIPE" which I distributed in September 1974.

Prior to 1971 any combination of loads or earthquakes had to be made by hand or by another program. In 1972 I released a summation procedure which enabled users to combine loads in accordance with B31.7 and Section III criteria. In 1973 the computation of fatigue usage factors was released, which included the cyclic effects due to various earthquake components. If these summation techniques were used, the user could input several transient (earthquake) loadings and combine these loadings, one by one, with a sustained loading (deadweight) to achieve a "worst case" stress calculation.

Modal Analysis by ADLPIPE During the Period 1977 to the Present

A new option was made available in ADLPIPE in February 1977 for the computation of earthquake response in accordance with Regulatory Guide 1.92, Revision 1, March 1976.

In addition, a post-processor has been developed which enables the user to make a number of combinations of directional earthquakes effects not included in Regulatory Guide 1.92.

Verification of ADLPIPE

Verification of ADLPIPE was undertaken in a series of fundamental checks. In important modifications a supporting document was prepared as an ADLPIPE reference. The verification procedure was as follows.

The thermal and deadweight loadings were checked by a Hovgaard Bend and hand calculated systems given in "Design of Piping Systems", M. W. Kellogg, Second Edition, 1956, and "Formulas of Stress and Strain", R.J. Roark, McGraw-Hill.

The dynamic analyses were checked by "Response of Structural Systems to Ground Shock", Shock and Structural Response, ASME, 1960, in "ADLPIPE Results of Model Given by Young (ADLPIPE Reference 4), and "Dynamic Behavior of a Foundation-Like Structure", Mechanical Independence Methods, ASME, 1958, in "Experimental Verification of ADLPIPE Mod 1" (ADLPIPE Reference 3).

The time history analysis was checked by a separate analytical solution of the problem given in "Analytical Methods of Vibrations," page 395, Leonard Meinovitch, "ADLPIPE Time History Response Compared with a Known Solution for a Heavily Damped System (ADLPIPE Reference 14). A second check was made using "Pressure Vessel and Piping 1972 Computer Progress Verification", ASME, 1972 (Problem 5).

The thermal transient analysis was verified by a separate analysis, "Transient Thermal Gradient Stresses", E. B. Branch, Heating, Piping and Air

Conditioning, Volume 43, 1971, pages 132-1[illegible], "ADLPIPE Thermal Transient Analysis" (Reference 15).

The computation of intra and inter modal moment component summation has been verified by a separate computer program for that purpose. A report "ADLPIPE Modal Response Combination for Closely Spaced Modes", is available as ADLPIPE reference 24.

Various calculation procedures required by ASME Section III were verified in ADLPIPE references 10, 11, and 18 entitled "ADLPIPE Computation of Bending Stress in Tees and Branch Connections, ASME Section III, Class 1 Piping", "ADLPIPE Computation of Resultant Moments for Section III Class 2 and 3 Stresses", and "ADLPIPE Stress Computation of Piping Components: A Comparison with Hand Calculations for ANSI B31 and ASME Section III."

In 1978 an independent third party review of ADLPIPE (Section III, Class 1) was performed "Verification of ADLPIPE, ASME Section III, Class 1 Piping Stress Program", Teledyne Engineering Services, Report No. TR-2884-1, August 11, 1978.

ADLPIPE Development Policy

The following policies have been in effect during the development of ADLPIPE:

1. The details of calculation processes are available to the public by free distribution of operating manuals and references. These are tabulated in Appendix I. Each major new feature of ADLPIPE is documented for user review.

2. Program listings are made available to licensees. Licensees are not restricted from making program changes.

3. ADLPIPE is periodically improved and updated and licensees are notified of the modifications at the time of the release of the modified version.

4. ADLPIPE is hand checked wherever possible. When this is not possible, ADLPIPE is checked by experimental results or the results of other calculation procedures. Every modification, large or small, is checked.

5. Special versions of ADLPIPE will be written to a licensee's specification. However, the version of ADLPIPE released to computer service bureaus generally does not have such special additions.

6. Old versions of ADLPIPE are not retained by Arthur D. Little, Inc. Instead, beginning in 1972, all new versions of ADLPIPE were backward integrated. The present version of ADLPIPE

maintains all past features which have been made available to the users during the period 1971 to 1979.

JW Dingwell

I. W. Dingwell
Arthur D. Little, Inc.
Cambridge, MA 02140
April 19, 1979

APPENDIX I

ADLPIPE VERSIONS AND DOCUMENTATION

Version	Documentation and Features
April 1968	ADLPIPE Thermal, Static, Dynamic Pipe Stress Analysis Operating Manual, undated.
April 1968	ADLPIPE Modification One: Thermal, Static, Dynamic Pipe Stress Analysis: Operating Manual, first version dated March 26, 1969. Features: Thermal, deadweight, external, acceleration and shock loads; single load stress analysis; code - B31.1 (1955).
August 1970	ADLPIPE....Static-Thermal-Dynamic Pipe Stress Analysis dated August 15, 1970 New Features: Code - B31.1 (1967); equations 9-13, B31.7
January 1971	ADLPIPE....Static-Thermal-Dynamic Pipe Stress Analysis dated January 15, 1971 New Features: Four modal summation techniques: maximum, maximum and square root sum of squares of remaining modes, square root sum of squares, absolute; square root sum of squares for stress calculations
July 1971 September 1971 November 1971 December 1971	ADLPIPE....Static-Thermal-Dynamic Pipe Stress Analysis dated April 1, 1971 New Features: Stress summary report, B31.7 for multiple loads
June 1972 July 1972 December 1972	ADLPIPE.....Static, Thermal, Dynamic Pipe Stress Analysis Input Preparation dated April 1, 1972 New Features: ASME Section III, Class 1 (1971), summary stress report of multiple loads; closely spaced modal summation References: 1. ADLPIPE Mathematical Analysis and Logical Procedure 2. Section III Sample Problem 3. Experimental Verification of ADLPIPE 4. ADLPIPE Results of Modal Given by D. Young 5. ADLPIPE Modification 1, Response to Ground Spectra 6. Development of Modal Participation Matrix for General Three Dimension Shock Input to Lumped Dynamic System
September 1973	ADLPIPE.....Static, Thermal, Dynamic Pipe Stress Analysis Input Preparation dated April 1973 New Features: English and Metric units; summary stress report, Section III Class 1 (1971); fatigue analysis (Class 1); graphical output: isometrics for input checking, dimensioned isometrics, static plots of deformed piping

References: 1. ADLPIPE Mathematical Analysis and Logical Procedure
2. Section III Sample Problem
3. Experimental Verification of ADLPIPE MOD 1
4. ADLPIPE Results of Model Given by D. Young
5. Generalized Piping System Response to Ground Shock Spectra
6. A Method of Computing Stress Range and Fatigue Damage in a Nuclear Piping System by W. B. Wright and E. C. Rodabaugh.

May 1974

ADLPIPE.....Static, Thermal, Dynamic Pipe Stress Analysis Input Preparation dated May 1974
New Features: Codes - B31.1 (1973): Section III, Class 1, 2, 3
New References: 7. Section III Sample Problem, Class 2, 3
8. ANSI B31.1 (1973) Sample Problem

April 1975
July 1975

ADLPIPE...Static and Dynamic Pipe Design and Stress Analysis: Input Preparation Manual dated January 1975
New Features: Revised input organization (geometry and execution decks)
New Reference: 9. ADLPIPE April 1975 Release

April 1976

ADLPIPE...Static and Dynamic Pipe Design and Stress Analysis: Input Preparation Manual dated January 1976
New Features: Section III Class 1, 2, 3 (1974); force time history dynamic analysis
New References: 5. Documentation of ADLPIPE for Static and Dynamic Loads and Stress Evaluation, September 1973.
6. A Method of Computing Stress Range and Fatigue Damage in a Nuclear Piping System, W. B. Wright and E. C. Rodabaugh, Nuclear Engineering and Design, 22 (1972).
7. Sample Stress Analysis of ASME Section III Nuclear Class 1 and Class 2, 3 Combined Piping System and ANSI B31.1 (1973) Piping System Computed by ADLPIPE.
8. ADLPIPE Skew Card Test Run, July 1975.
9. ADLPIPE April 1976 Release.
10. ADLPIPE Computation of Bending Stress in Tees and Branch Connections, ASME Section III, Class 1 Piping, July 1975.
11. ADLPIPE Computation of Resultant Moments for Section III Class 2 and 3 Stresses July 1975.
12. ADLPIPE Detection and Reduction of Numerical Round-off Error with Springs and Stiff Members, July 1975.

Royalty Agreements

COMPANY	ADDENDUM	EFFECTIVE DATE	EXPIRATION DATE
Black & Veatch		10/04/74	perpetual
Blaw-Knox		10/04/67	perpetual
Brown & Root		11/07/75	perpetual
Burns & Roe		7/22/77	perpetual automatic extension
	P.O.	8/19/77	
	1	PENDING	
Comision Federal de Electricidad		7/ /74	perpetual
	1	2/23/76	
Framatone		11/29/72	perpetual
	1	11/16/75	
	2	7/20/76	
Gibbs & Hill		2/20/70	perpetual
		7/06/72	perpetual
	1	11/01/78	automatic extension
M. W. Kellogg Company		5/12/70	perpetual
Charles T. Main, Inc.		11/19/73	11/19/76
	P.O.	10/26/76	perpetual
Montreal Engineering Company & Monenco Computing Service Ltd.		4/18/72	perpetual
	1	6/25/75	
Northeast Utilities Service Company		?	automatic extension
		= 1/01/77	
Power Piping Co.		5/12/70	perpetual

Royalty Agreements (cont)

COMPANY	ADDENDUM	EFFECTIVE DATE	EXPIRATION DATE
Sener Ingeniera y Sistemas, S.A.		2/01/72	perpetual
	1	10/01/73	
	2	6/16/75	
	3	5/01/77	
	4	11/01/78	
United Engineers & Constructors		5/20/70	perpetual
Westinghouse Electric Corporation		11/27/67	perpetual

V. MODIFICATION I: RESPONSE TO GROUND SHOCK SPECTRA

The basic approach to be used in computing the response of piping systems to ground shock inputs in terms of displacement (or velocity or acceleration) spectra consists of generating the dynamic properties of the system and applying a modal superposition method (or normal mode method) to define the structural response to the shock inputs. The formulation in terms of normal modes follows generally the form discussed by Young.[1] As formulated in this reference, the contributions from the individual normal modes are defined in terms of a modal participation factor which depends upon the normal shape (eigenvector) and the distribution of the load over the structure. This formulation is applicable, however, to systems excited by one-dimensional shock only, i.e., with the inertial elements restricted to motions in a plane. For the general three-dimensional shock input and response case, the contributions of the normal modes can be shown to be defined in a modal participation matrix.

A description of the steps leading to the determination of the response due to ground shock is given in the following paragraphs.

A. Calculation of "Reduced" Stiffness Matrix

In order to define the normal modes of the piping systems, a flexibility or stiffness matrix relating forces and deflections at the mass points in the system must be generated. Following the procedure in ADLPIPE, a network stiffness matrix is first formed as an N by N array for a system of N network points. (Each of these N x N "terms" are 6 x 6 subsets.) The numbering of the network points is carried out in the following priority: first, the mass points; second, the interior branch points; and finally, the anchor points. The stiffness matrix,

1. Young, Dana "Response of Structural Systems to Ground Shock", Shock and Structural Response, American Society of Mechanical Engineers, N. Y., 1960.

thus formed will be ordered as indicated below:

$$\left[\begin{array}{c|c|c} A & B & C \\ D & E & F \\ \hline G & H & I \end{array}\right]$$

A	mass points sub-matrix
B, D, E	interior points sub-matrices
C, F, G, H, I	anchor points sub-matrices

As shown, the matrix is partitioned into the three categories of network points. The formation of the complete matrix is carried out by ADLPIPE.

The rows and columns corresponding to anchor points are now deleted from the stiffness matrix, leaving a matrix characterizing mass points and interior branch points only.

$$\left[\begin{array}{c|c} A & B \\ \hline D & E \end{array}\right] * \begin{bmatrix} \Delta_0 \\ \Delta_1 \end{bmatrix} = \begin{bmatrix} F_0 \\ F_1 \end{bmatrix}$$

Δ_0 represents deflections at the mass points, and Δ_1 represents deflections at the interior branch points. Similarly, F_0 represents loads at the mass points, and F_1 loads at the interior branch points. In the case of free vibration, the loads F_0 are inertial loads due to the mass points and the loads F_1 are zero since interior network points are not loaded. The equations then become

$$A \cdot \Delta_0 + B \cdot \Delta_1 = F_0$$

$$D \cdot \Delta_0 + E \cdot \Delta_1 = 0$$

From the second equation, $\Delta_1 = -E^{-1} D \Delta_0$. Substituting into the first equation

$$(A - B \cdot E^{-1} D) \Delta_0 = F_0$$

This results in a "reduced" stiffness matrix, relating the forces and deflections at mass points. This matrix is an n x n array where n is the number of mass points. For inertia loads, this matrix equation may be written as

$$K_R \cdot \Delta_0 = \omega^2 M \Delta_0$$

where $$K_R = (A - B\ E^{-1}\ D)$$

B. Calculation of Normal Modes

The eigenvectors, Δ_{0_n}, and the eigenvalues, ω_n, for each of the normal modes are computed by solving the matrix equation

$$K_R \cdot \Delta_0 = \omega^2 M \Delta_0$$

for each of its n characteristic solutions. This equation may be solved by iterative procedure when put into the form

$$A\ V_n = \omega_n^2 \cdot V_n$$

This transformation is performed by defining

$$M = M^{1/2}\ M^{1/2}$$

and $$V_n = M^{1/2}\ \Delta_{0_n}$$

thus defining the matrix A as

$$A = M^{-1/2}\ K_R\ M^{1/2}$$

It assures that the iteration will converge and have real and positive eigenvalues[2].

2. Wada, B. Stiffness Matrix Structural Analysis, Jet Propulsion Laboratory, Technical Report No. 32-774, October 31, 1965.

With the matrix equation in this form, the iterative process will converge most readily on the eigenvalue having the largest magnitude. For ground shock response applications, it is more desirable for the process to converge most readily to the smallest eigenvalue. Consequently, the matrix equations are put in the inverted form

$$C\,V_n = \lambda_n V_n$$

where $$C = A^{-1} \text{ and } \lambda_n = 1/\omega_n^2$$

and application of an iterative method, such as the Stodola method, will produce the successive modal frequencies (eigenvalues) and modal columns (eigenvectors) of a system in ascending order.

An alternative solution technique, which has been utilized in ADLPIPE MOD 1 is the Jacobi method[3]. In this procedure, all of the eigenvalues and eigenvectors are produced simultaneously with equal accuracy. This method may, therefore, employ the matrix equation in either form (i.e., with eigenvalues $1/\omega_n^2$ or ω_n). In MOD 1, the second form, in terms of λ_n, has been used. The modal frequencies are stored in a "frequency vector", and the modal columns are stored in modified form as columns in a "modal matrix". The modal columns are modified by first converting the V_n back to modal deflections Δ_0 and then by normalizing the column. Each of the set within a modal column, ϕ_{in}, now represents a normalized deflection of mass i in mode n.

C. Calculation of Equivalent Static Deflections

As indicated in the appendix, the modal amplitude q_n is shown to be given by the expression

$$q_n = \sum_{\ell} \Phi_{n\ell} (D_\ell)_n$$

3. Greenstadt, J. "The Determination of the Characteristic Roots of a Matrix by the Jacobi Method", Chapter 7 of Mathematical Methods for Digital Computers, John Wiley, New York, 1959.

where $\Upsilon_{n\ell}$ is the modal participation matrix and $(D_\ell)_n$ is the shock input displacement for each coordinate and for each mode. This general three-dimensional form reduces to the simpler formulation

$$q_n = \gamma_n D_n$$

in the case that the input shock motion at the base is the same in every coordinate. It is this latter form which is developed by Young[(1)]. For this one-dimensional case as discussed in Reference 1, the modal participation factor is defined for each mode, while for the general three-dimensional case, the modal participation is defined for each mass for each mode, and thus is in a square array form rather than in a linear array form.

The amplitudes $(D_\ell)_n$ are obtained from the given input shock spectra (e.g., Housner spectra for earthquake loadings). In these spectra, the amplitudes are defined by the modal frequency and by the coordinate axis. For each value of ω_n, therefore, and for each coordinate axis in which there is a prescribed input spectra, we have a value of $(D_\ell)_n$. The modal amplitude q_n is then evaluated as the scalar summation of the products of the nth vector of the modal participation matrix and the spectra amplitude $(D_\ell)_n$, or, as given previously,

$$q_n = \sum_\ell \Upsilon_{n\ell} \, (D_\ell)_n$$

The modal amplitudes are now converted to amplitudes in the original coordinate system by the relation

$$u_i = \Phi_{in} \, q_n$$

1. Young, Dana "Response of Structural Systems to Ground Shock", Shock and Structural Response, American Society of Mechanical Engineers, N. Y., 1960.

This now provides a set of displacements, u_i, for each of the n modes. These individual sets of displacements can now be applied to the system as equivalent static deflections. The corresponding network forces are obtained by the usual procedures of ADLPIPE. It should be noted that the stiffness matrix to be used for this procedure must be that resulting when the rows and columns corresponding to anchor points are deleted, i.e.,

$$\begin{bmatrix} A & B \\ D & E \end{bmatrix}$$

The reduced stiffness matrix $(A-BE^{-1}D)$ cannot be used, since interior points (branch points) must be considered in the process of transferring interior loads and deflections from point to point.

ADLPIPE utilizes the network force sets to generate stresses for each mode. The upper bound for the stress levels at any point in the system is given by the absolute summation of the stresses generated for each mode. Such a summation assumes that the contributions from each mode reach their maximum value at the point in question at the same time. Other methods of summation may be used, of course, depending on the degree of conservatism desired in the analysis. A suggested alternative, for example, might be the sum of the contribution of the fundamental mode and the rms summation of the higher mode.

(where ϕ_{in} are the modal columns of the modal matrix) and the corresponding transformations between $\dot{u}_i$ and $\dot{q}_n$, $\ddot{u}_i$ and $\ddot{q}_n$, $\dot{s}_i$ and $\dot{p}_n$, and $\ddot{s}_i$ and $\ddot{p}_n$. We further define the generalized inertia by

$$M_{k\ell} = \sum_{ij} m_{ij} \phi_{ik} \phi_{j\ell}$$

Because of orthogonality, the generalized inertia matrix is a diagonal matrix, and hence may be written as

$$M_{k\ell} = M_k \delta_{k\ell}$$

where $\delta_{k\ell}$ = Kronecker delta.

Because of the symmetry of the inertia matrix, m_{ij}, the bilinear form $\sum_{ij} m_{ij} \dot{u}_j \dot{s}_i$ becomes

$$\sum_{ijk\ell} m_{ij} \phi_{ik} \dot{q}_k \phi_{j\ell} \dot{p}_\ell = \sum_{k\ell} M_k \delta_{k\ell} \dot{q}_k \dot{p}_\ell = \sum_k M_k \dot{q}_k \dot{p}_k$$

and the bilinear term $\sum_{ij} m_{ij} \dot{u}_i \dot{s}_j$ also becomes $\sum_k m_k \dot{q}_k \dot{p}_k$

Similarly, the quadratic forms $\sum m_{ij} \dot{u}_i \dot{u}_j$ and $\sum m_{ij} \dot{s}_i \dot{s}_j$ become

$$\sum_k M_k (\dot{q}_k)^2 \text{ and } \sum_k M_k (\dot{p}_k)^2.$$

The kinetic energy may, therefore, be written as

$$T = \frac{1}{2} \sum_k [M_k \dot{q}_k^2 + 2M_k \dot{q}_k \dot{p}_k + M_k \dot{p}_k^2]$$

From the definition of normal modes, we have $\sum_j (k_{ij} - \omega_k^2 m_{ij})\phi_j = 0$ where ϕ_j is an eigenvector (modal column) and ω_k is the corresponding eigenvalue.

For the nth modal column and the nth eigenvalue, this becomes

$$\sum_j [k_{ij} - (\omega_n)^2 m_{ij}] \phi_{jn} = 0$$

or

$$\sum_j k_{ij} \phi_{jn} = (\omega_n)^2 \sum_j m_{ij} \phi_{jn}$$

$$\sum_{ij} k_{ij} \phi_{jn} \phi_{is} = (\omega_n)^2 \sum_{ij} m_{ij} \phi_{jn} \phi_{is} = \omega_n^2 M_n \delta_{ns}$$

Now the potential energy is given by $V = \frac{1}{2} \sum_{ij} k_{ij} u_i u_j$

$$= \frac{1}{2} \sum_{ijns} k_{ij} \phi_{in} q_n \phi_{is} q_s$$

$$= \frac{1}{2} \sum_{ns} \left(\sum_{ij} k_{ij} \phi_{in} \phi_{js} q_n q_s \right)$$

$$= \frac{1}{2} \sum_{ns} \omega_n^2 m_n \delta_{ns} q_n q_s$$

$$= \frac{1}{2} \sum_n \omega_n^2 m_n q_n^2$$

The appropriate form of Lagrange's equation is

$$\frac{d}{dt} \left(\frac{\partial T}{\partial \dot{q}_k} \right) + \frac{\partial V}{\partial q_k} = 0$$

which gives the equations of motion, $\ddot{q}_k + \omega_k^2 q_k = P_k$

The solution of this equation for the modal amplitude q_k is

$$q_k (t) = \frac{1}{\omega_k} \int_0^t P_k (T) \sin \omega_k (t-T) dT$$

From the transformation equation, the vector p_k is related to the vector g_k which describes the accelerations of the base of the system. This relation is $p_k = \sum_\ell \phi^{-1}_{k\ell} g_\ell$ where $\sum_\ell \phi^{-1}_{k\ell} \phi_{\ell m} = \delta_{km}$, the identity matrix. Hence, $q_k(t) = \frac{1}{\omega_k} \sum_\ell \int_0^t \phi^{-1}_{k\ell} g_\ell(T) \sin \omega_k (t-T)dT$.

If $[R_e(t)]_k = \frac{1}{\omega_k} \int_0^t g_\ell(t) \sin \omega_k (t-T)dT$ and the shock input displacement $(D_\ell)_k = |[R_e(t)]_k|_{max}$, the input spectra are defined as the maximum modulus of the response value $R_\ell(t)$. The modal amplitudes now become $q_k \le \sum_\ell \phi^{-1}_{k\ell} (D_\ell)_k$, or $q_k \le \sum_\ell \Psi_{k\ell} (D_\ell)_k$ where $\Psi_{k\ell}$ is an element of the modal participation matrix for mode k, mass point ℓ.

In the special case in which all of the elements of g_ℓ are the same and all of the $(D_\ell)_k$ are the same, this expression reduces to

$$|q_k|_{max} = D_k \sum_\ell \Psi_{k\ell}$$

This is an alternative expression for Young's result

$$|q_k|_{max} = D_k \, \gamma_k$$

since it can be shown that the modal participation factor, γ_k, is equal to $\sum_\ell \Psi_{k\ell}$. The modal participation factor is appropriate, however, only for the special case when all of the elements of the base acceleration vectors g_ℓ are the same. This is not the case, of course, in three-dimensional shock motions with different shock inputs (spectra) in the various coordinate axes.

DYNAMIC STRESS ANALYSIS BY ADLPIPE

by

I. W. Dingwell
Arthur D. Little, Inc.

On page 5 of the reference* an expression for a set of displacements is developed for each mass degree of freedom and for each mode:

$$X_i = \phi_{in} q_n$$

These displacements are developed from the normalized set of displacements q, as transformed by the modal matrix ϕ_{in} for mass direction i and mode n.

The displacements, X_i, represent the zero to peak displacement of each mass degree of freedom when subjected to a shock loading which is described by a (displacement/velocity/acceleration vs. frequency) response spectra. The displacements have a consistent set of algebraic signs which define the mode shape of the deflected piping system. Reversing the signs of the displacements gives the opposite peak modal deflections of the piping system.

From this set of modal displacements, X_i, the displacements of the non-mass points are calculated. There are two types of non-mass points: a) non-mass network points, and b) interior points within a pipe section. Since ADLPIPE uses a transfer matrix technique for combining several pipe elements in series to formulate the stiffness of the section (a section is a series of connected elements), the non-mass network points are calculated first, then those deflections are utilized to calculate reactions at the network points. Finally, internal forces, moments, and deflections are calculated by transferring the initial boundary conditions across each member in a section.

Thus, for each mode, a set of moments is calculated:

$$M_{kjn}$$

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*Generalized Piping System Response to Ground Shock Spectra by Irving W. Dingwell, Arthur D. Little, Inc., Cambridge, Massachusetts.

APPENDIX IV

DEVELOPMENT OF MODAL PARTICIPATION MATRIX FOR GENERAL THREE-DIMENSION SHOCK INPUT TO LUMPED DYNAMIC SYSTEM*

The development of the modal participation factor in the analysis of the response of a lumped dynamic system to a one-dimensional shock input is carried out by Young in Reference 1 by application of Lagrange's equation with the system kinetic and potential energies expressed in terms of normal coordinates. In this appendix, this development is extended to include the general loading case in which different shock inputs are allowed in each of the system coordinate axes. The terminology utilized by Young has been followed to the extent possible.

For the lumped system defined by the symmetric inertia matrix m_{ij} and by the stiffness matrix k_{ij}, let u_i be the elastic displacement in the i^{th} coordinate, and let $u_i + s_i$ be the absolute displacement. The elements u_i and s_i are, in the general case, six element vectors for each mass point.

The kinetic energy T of the system is given by

$$T = \frac{1}{2} \sum_{ij} m_{ij} (\dot{u}_j + \dot{s}_j)(\dot{u}_i + \dot{s}_i)$$

$$= \frac{1}{2} \sum_{ij} (m_{ij} \dot{u}_i \dot{u}_j + m_{ij} \dot{u}_j \dot{s}_i + m_{ij} \dot{u}_i \dot{s}_j + m_{ij} \dot{s}_i \dot{s}_j)$$

The potential energy V is given by

$$V = \frac{1}{2} \sum_{ij} k_{ij} u_i u_j$$

We introduce the normal coordinates $q_n(t)$ and $p_n(t)$ by the linear transformations:

$$u_i(t) = \sum_n \phi_{in} q_n(t)$$

$$s_i(t) = \sum_n \phi_{in} p_n(t)$$

*September 30, 1974

where k = orthogonal axis (X, Y, Z coordinate)

j = earthquake direction (X, Y, Z axis response spectra)

n = mode

With a normal mode analysis, all coupling and phase relationships between modes are unknown. However, since these moments have algebraic signs and refer to a consistent position on the piping surface, the question of how to sum the modal moments arises.

The present version of ADLPIPE assumes that earth motion is oriented along a single vector and is composed by a spectra with components in the three orthogonal axes. Therefore, in a single mode, the piping responds "in phase"

$$M_{nk} = \sum_{j=1}^{3} M_{kjn} \qquad \text{(Equation 1)}$$

and the algebraic sum is taken of the motion which results from the single earthquake. The response is independent of axis orientation.

Since there is no phase relationship between modes, a mean summation must be taken. The present version uses the square root sum of squares.

$$M_k = \left(\sum_{n=1}^{n_{max}} \left(\sum_{j=1}^{3} M_{kjn} \right)^2 \right)^{1/2} \qquad \text{(Equation 2)}$$

There is an alternative technique which implies that closely spaced modes are coupled and are taken to be in phase. Therefore, when that occurs, the square root sum of squares is taken of the absolute sum of the closely spaced modal moments.

For instance, modes 1 and 2 are closely spaced

$$M_k = \left(\left(\sum_{j=1}^{3} |M_{kj1}| + \sum_{j=1}^{3} |M_{kj2}| \right)^2 + \sum_{j=1}^{3} (M_{kj3})^2 \right)^{1/2} \qquad \text{(Equation 3)}$$

The test for the closely spaced modes is:

if $\frac{(f_2 - f_1)}{f_1} < k$, then the bandwidth factor (k) for these modes will cause the program to form an absolute sum. (This type summation must be requested of the program by the analyst. At present, the factor K in percent is entered in the Z2 field on the SHOCK card. If Z2 = 0., then equation 2 is utilized.)

ALTERNATIVE SOLUTIONS

A conservative assumption is that the vibratory energy in an earthquake is random and the component moments along each axis are independent of one another. Realistically, the earthquake acts as three different earthquakes, with the axis orientation a variable. Therefore, since phase relationships are unknown, a mean solution is taken independently for each shock direction. In mode n,

$$M_{nk} = \left(\sum_{j=1}^{3} (M_{kjn})^2\right)^{1/2} \qquad \text{(Equation 4)}$$

Following the square root summation for the modes n to n_{max}

$$M_k = \left(\sum_{n=1}^{n_{max}} \sum_{j=1}^{3} |M_{kjm}|\right. \qquad \text{(Equation 5)}$$

Since the absolute sum is overly conservative, an alternative is to take the maximum modal response plus the square root sum of the square of the remaining moments.

$$M_k = \left(\sum_{j=1}^{3} |M_{kjm}|\right)_{max} + \left(\sum_{n=1}^{n_{max}-1} \cdot \left(\sum_{j=1}^{3} M_{kjm}\right)^2\right)^{1/2} \qquad \text{(Equation 6)}$$

Each of these alternative solution summation schemes or variations thereon can be inserted, upon request, into the ADLPIPE program.

The resulting stress analysis is dependent on the summation of the modal moments. The example given here is not a statistical mean but certainly indicates that the present version of ADLPIPE is unconservative but realistic. As a consensus is reached, other summation techniques will be introduced.

Numerical Example

At point zero in ASME Section III Sample Problem (Class 1, Class 2)

		Shock Dir.	M_x	M_y	M_z
mode n = 1		x	-47467	-1297	67221
		y	-153624	-4199	217557
		z	-27185	-743	38498
algebraic sum	(Equation 1)		-228276	-6239	323276
SRSS	(Equation 4)		163071	4457	230936
mode n = 2		x	-27343	-5128	2882
		y	159117	29843	-16774
		z	-851446	-159690	89758
algebraic sum	(Equation 1)		-719672	-134975	75866
SRSS	(Equation 4)		866617	162535	91357
mode n = 3		x	101890	3195	1426756
		y	-29914	-938	-418883
		z	-8862	-278	-124098
Algebraic sum	(Equation 1)		63114	1979	883725
SRSS	(Equation 4)		106559	3341	1492144

TOTAL moment computed by		M_x	M_y	M_z	$(M_x^2 + m_y^2 + m_z^2)^{1/2}$	RATIO
SRSS of alg. sum	(Equation 2)	757641	135133	944098	1218031	1.0
SRSS of SRSS	(Equation 4)	828241	162630	1512670	1761701	1.44
Absolute sum of alg. sum	(Equation 1)	1011062	143193	1282867	1639664	1.34
Absolute sum of SRSS	(Equation 3)	1136247	170333	1814437	2147615	1.26
Max. + SRSS of alg. sum	(Equation 1)	956512	141520	1215783	1553406	[illegible]
Max. + SRSS of SRSS	(Equation 6)	1061416	168105	1740499	2045531	

QUESTION 13: We understand that certain shortcomings might have been experienced with respect to informing all members of the Commission of the development leading to orders for the shutdown. Please clarify this matter and indicate any stops taken to improve communications to the entire Commission.

ANSWER:

The problems encountered regarding informing the Commissioners in a timely manner of important matters has been addressed and rectified. The Director of each NRC Office has been instructed that each Commissioner is to be promptly and individually notified in such situations. It is our intention that this occurrence will not be repeated.

Regarding the particular instance at hand, the five plant shutdown, the following information is supplied for clarification. On Friday, March 9, the staff was aware that an area of concern existed regarding the seismic design of certain nuclear power plants. Because of the preliminary nature of our information, the potential severity of the problem was not identified at that time. However, the information available to the staff had been communicated to Chairman Hendrie. On the following weekend, NRC staff members went to Stone and Webster offices in Boston for further information. It was at this time that the potential magnitude of the problem was fully recognized and initial steps toward issuing the show cause orders were taken. Because of the short time period between our recognizing the need for the orders and their issuance and because much of what precipitated the final actions occurred over the weekend, the full Commission was not kept properly informed. The response to Question 9 provides a detailed chronology of these events.

United States Senate

COMMITTEE ON ENVIRONMENT AND PUBLIC WORKS
WASHINGTON, D.C. 20510

June 26, 1979

Mr. Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Chairman:

Please provide responses to the attached follow-up questions to the Subcommittee on Nuclear Regulation's Hearing on March 16 regarding the shutdown of five nuclear power plants because of an error in the analyses of the seismic design. So that the record may be completed, we would appreciate receiving your responses by July 20.

Sincerely,

Gary Hart

Gary Hart
Chairman, Subcommittee on
Nuclear Regulation

ENCL.

FIVE PLANT SHUTDOWN

1. When performing cost/benefit analyses of alternatives in NEPA reviews, how does NRC factor into those analyses costs such as those entailed in shutdowns (whether voluntary or by order or license conditions) of reactors because of safety concerns?

2. How has NRC assured that the codes being used in the reanalyses of seismic design produce valid results?

3. What steps have been taken to assure other computer codes currently being used for reactor designs do not contain errors?

4. Please list each reactor which has been found since March 13, including the five reactors which were the subject of the hearing, to have had an error in the seismic analyses of plant design. In your response, please include:

(a) whether the reactor was shutdown because of the error;

(b) whether the shutdown was voluntary or by order;

(c) the systems involved;

(d) whether the systems are safety related or non-safety related, and

(e) the resulting corrective measures if any.

5. (a) What technical standards/methods are being used to determine the adequacy of design for seismic events - those existing at the time the 5 plants were licensed or those in existance at this time? If the former, please describe:

(b) the differences;

(c) the rationale for not applying modern standards, and

(d) a brief assessment of the relation between the existing seismic designs for the 5 plants and the existing standards.

6. (a) How do the perceived risks associated with the error in the seismic design of the 5 plants compare with those associated with the Babcock and Wilcox plants during the first five weeks following the accident at Three Mile Island?

(b) What factors led to the shutdown of all of the former within a few days of learning of the shortcomings, while some Babcock and Wilcox plants never were shutdown?

7. (a) What are the recurrence frequency and magnitude of the design basis and operating basis earthquakes at each of the 5 plants?

   (b) Based on the reanalyses using acceptable procedures, what are the recurrence frequency and magnitude of the earthquake that would have resulted in stresses above the allowable limit prior to any plant modifications.

8. What are the estimated costs of the shutdowns of the 5 plants in terms of dollars and barrels of oil? The underlying assumptions should be stated.

9. In the March 16 hearing, Mr. Denton remarked that much credit for bringing the computer error to his attention goes to the diligence of an NRC inspector who pursued the discrepancy in the results of the old and new codes. Please provide the particulars in a chronology of the surfacing of the discrepancy and an assessment of the reasons for any delays.

10. Please provide available information on the recent earthquake that occurred in the vicinity of the Maine Yankee plant. How does it compare with the operating basis earthquake.

11. One of the plants ordered shut down is the Surry Plant which served as the model PWR for the Reactor Safety Study (RSS). The RSS included an extensive design adequacy study.

   (A) What was the finding of the study team with respect to seismic design of Surry?

   (b) What are the ramifications with respect to future quantitative risk assessments ?

12. Please list all nuclear powerplants that have been exported from the United States that were designed with the aid of the erroneous computer code involved in the five plant shutdowns.

13. We understand that certain shortcomings might have been experienced with respect to informing all members of the Commission of the development leading to orders for the shutdown. Please clarify this matter and indicate any steps taken to improve communications to the entire Commission.