William J. Cahill, Jr.
Vice President

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

February 15, 1978

Director of Nuclear Reactor Regulation
ATTN: Mr. Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Reid:

Forwarded herewith for your information is a copy of the Tenth Quarterly Report for the Seismic Monitoring Program for Indian Point covering the months of September 1977 through November 1977.

Very truly yours,

William J. Cahill

William J. Cahill, Jr.
Vice President

Copy to: Mr. George T. Berry
General Manager and Chief Engineer
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

WC/nvg

docket

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

February 10, 1978

50-247✓
50-286

To All PWR Facility Licensees

Gentlemen:

By letter dated December 9, 1977, copy enclosed, we requested you and all other PWR facility licensees to complete and submit a questionnaire on steam generator operating history that was enclosed. The letter stated that the request for information was approved by GAO under a blanket clearance. Questions have been raised about the appropriateness of this request for information in light of the Federal Reports Act and about the referenced GAO blanket clearance. These questions have been discussed with representatives of GAO and it was determined that this clarifying letter should be sent to each recipient of our original letter. GAO has agreed that this request properly fits under the GAO blanket clearance for reports concerning possible generic problems and the applicable GAO clearance number should have been R0072 rather than R0071.

The request for additional information was prompted by the continuing degradation of tubes in all three vendors' steam generators. Such degradation is an important safety concern of the NRC because such tubes form part of the primary coolant pressure boundary. Several forms of degradation that have been observed in steam generators in recent months have included the wastage of tubes at Palisades and other facilities, stress corrosion at Ginna and other facilities, vibration cracking and "dinging" of tubes at the Oconee (B&W) facilities, antivibration bar fretting at San Onofre, and "denting" of tubes and associated support plate "hourglassing" and cracking at Surry, Turkey Point and about 15 other CE and W facilities. These events have resulted in many shutdowns of nuclear power stations and the safety significance of certain of these events have prompted the NRC to issue safety Orders. It is this need for important safety information that has dictated this request for additional information.

Dupe of ~~8004020573~~

ma4

Our previous letter acknowledged that selected portions of the information being requested may already be available to the NRC, but not in a convenient format which is readily accessible. We therefore requested that you assist us by returning a single completed copy of the enclosed questionnaire. We would like to clarify that an acceptable response to any item in the questionnaire would be to provide specific reference to any information previously submitted to the NRC, by an original response, or any combination thereof, whichever and for whatever reasons you elect to use.

Our previous letter further requested that you submit any changes or additions to your initial submittal to reflect the future operating experience with your steam generators. This would enable us to maintain the information current, which, as we stated, we will periodically publish and send copies to all participants. As we indicated, this would enable the NRC, you and others to draw from the operating experience of the entire nuclear industry on an ongoing basis when making safety and other decisions concerning steam generators in PWR plants. We are planning to prepare a submission to GAO for clearance of a request for reporting information regarding changes or additions to your initial submittal under this request.

We hope that the need for this clarification caused you no inconvenience. Because of the problems discussed above, we are extending the date for submitting the requested information to March 1, 1978.

Sincerely,

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Enclosures:
1. Letter dtd. 12/9/77 to PWR Licensees
2. Questionnaire

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

December 9, 1977

TO ALL PWR FACILITY LICENSEES

Gentlemen:

The NRC staff has recently been engaged in a series of discussions with reactor vendors, EPRI, and the Steam Generator Owners Groups concerning steam generator operational problems. Central to these discussions is an accurate assessment of operational conditions and experiences as well as the programs aimed towards the resolution of these problems.

In order to ensure that both the NRC and the nuclear industry have available a comprehensive collection of operating data for steam generators to permit informed, timely decisions and actions, DOR is establishing a steam generator information system. The system will collect appropriate information from all PWR licensees which will periodically be published. It is presently anticipated that the initial publication of information will be in the early part of 1978. You will be sent a copy of this and all future publications.

This information system will enable the NRC and each Licensee to draw from the operational experiences of the entire nuclear industry when making any decisions concerning steam generators. This should result in both safety and economic benefits.

Enclosed is a questionnaire which we request that you complete for each of your operating PWR units. We believe that the questionnaire is self explanatory, however, if questions arise or any clarifications are necessary, please do not hesitate to contact your NRC Project Manager. Please include with your response any diagrams you may have available which illustrate the tube plugging and/or the tube degradation patterns.

To enable us to maintain the information current, you are further requested to submit in the same format indicated by the questionnaire, any changes or additions to your initial submittal to reflect the future operating experience with your steam generators, i.e., the results of future steam generator inspections. The questionnaire should be completed to the extent applicable and appropriate at this time, i.e. regardless of operating experience.

The information being requested is quite extensive and will require a diligent effort on your part and ours to assure accurate and timely completion. Also, we realize that parts of the information may already be available to the NRC, but not in a convenient format which is readily accessible. Therefore, we request that you assist us by returning a single completed copy of the enclosed questionnaire to the Director of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, within 60 days of receipt of this letter. Please include any comments or suggestions for improving this information system which you may have.

This request for generic information was approved by GAO under a blanket clearance number R0071; this clearance expires September 30, 1978.

Sincerely,

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Enclosure:
Steam Generator
Operating History
Questionnaire

cc w/enclosure:
See next page

ENCLOSURE
STEAM GENERATOR OPERATING
HISTORY QUESTIONNAIRE

<u>NOTE</u>: All percentages should be reported to four significant figures.

I. BASIC PLANT INFORMATION

Plant:

Startup Date:

Utility:

Plant Location:

Thermal Power Level:

Nuclear Steam Supply System (NSSS) Supplier:

Number of Loops:

Steam Generator Supplier, Model No. and Type:

Number of Tubes Per Generator:

Tube Size and Material:

II. STEAM GENERATOR OPERATING CONDITIONS

<u>Normal Operation</u>

Temperature:

Flow Rate: Allowable Leakage Rate:

Primary Pressure:

Secondary Pressure:

<u>Accidents</u>

Design Base LOCA Max. Delta-P:

Main Steam Line Break (MSLB) Max. Delta-P:

III. STEAM GENERATOR SUPPORT PLATE INFORMATION

Material:

Design Type:

Design Code:

Dimensions:

Flow Rate:

Tube Hole Dimensions:

Flow Hole Dimensions:

IV. STEAM GENERATOR BLOWDOWN INFORMATION

Frequency of Blowdown:

Normal Blowdown Rate:

Blowdown Rate w/Condenser Leakage:

Chemical Analysis Results

Results	Parameter Control Limits

V. WATER CHEMISTRY INFORMATION

Secondary Water

Type of Treatment and Effective Full Power (EFP) Months of Operation:

Typical Chemistry or Impurity Limits:

Feedwater

Typical Chemistry or Impurity Limits:

Condenser Cooling Water

Typical Chemistry or Impurity Limits:

Demineralizers - Type:

Cooling Tower (open cycle, closed cycle or none):

VI. TURBINE STOP VALVE TESTING (applicable to Babcock & Wilcox (B&W) S.G. only)

Frequency of Testing

Actual:

Manufacturer Recommendation:

Power Level At Which Testing Is Conducted

Actual:

Manufacturer Recommendation:

Testing Procedures (Stroke length, stroke rate, etc.)

Actual:

Manufacturer Recommendation:

VII. STEAM GENERATOR TUBE DEGRADATION HISTORY

(The following is to be repeated for each scheduled ISI)

Inservice Inspection (ISI) Date:

Number of EFP Days of Operation Since Last Inspection:

(The following is to be repeated for each steam generator)

Steam Generator Number:

Percentage of Tubes Inspected At This ISI:

Percentage of Tubes Inspected At This ISI That Had Been Inspected At The Previous Scheduled ISI:

Percentage of Tubes Plugged Prior to This ISI:

Percentage of Tubes Plugged At This ISI:

Percentage of Tubes Plugged That Did Not Exceed Degradation Limits:

Percentage of Tubes Plugged As A Result of Exceedance of Degradation Limits:

Sludge Layer Material Chemical Analysis Results:

Sludge Lancing (date):

Ave. Height of Sludge Before Lancing:

Ave. Height of Sludge After Lancing:

Replacement, Retubing or Other Remedial Action Considered: (Briefly Specify Details)

Support Plate Hourglassing:

Support Plate Islanding:

Tube Metalurgical Exam Results:

Fretting or Vibration in U-Bend Area (not applicable to B&W S.G.) AS OF (4)

Percentage of Tubes Plugged	Other Preventive Measures

Wastage/Cavitation Erosion AS OF (4)

Hot Leg: (Repeat this information for the cold leg on Combustion Engineering (C.E.) and Westinghouse (W) S.G.)

Area of Tube Bundle (1)	a	b	c	d	e
% of Tubes Affected by Wastage/Cavitation Erosion					
% of Tubes Plugged Due to Exceedance of Allowable Limit (2)					
% of Tubes Plugged That Did not Exceed Degradation Limit					
Location Above Tube Sheet (3)					
Max. Wastage/Cavitation Erosion Rate for Any Single Tube (Tube Circum. Ave) (Mills/Month)					
Max. Wastage/Cavitation Erosion in Any Single Unplugged Tube (Tube Circum. Ave) (Mills)					

Cracking AS OF (4)

Caustic Stress Corrosion Induced in C.E. and W S.G.

Flow Induced Vibration Caused in B&W S.G.

Cracking (Con't)

Hot Leg: (Repeat this information for the cold leg on C.E. and W S.G.)

Area of Tube Bundle (1)	a	b	c	d	e
% of Tubes Affected By Cracking					
% of Tubes Plugged Due to Cracking					
% of Tubes Plugged That Did Not Exceed Degradation Limit					
Location Above Tube Sheet (3)					
Rate of Leakage From Leaking Cracks (gpm)					

Denting (Not applicable to B&W S.G.) AS OF (4)

Hot Leg: (Repeat this information for the cold leg on C.E. and W S.G.)

Area of Tube Bundle (1)	a	b	c	d	e
% of Tubes Affected by Denting					
% of Tubes Plugged Due to Exceedance of Allowable Limit (2)					
% of Tubes Plugged That Did Not Exceed Degradation Limit					
Rate of Leakage From Leaking Dents (gpm)					
Max. Denting Rate for Any Single Tube (Tube Circum. Ave) (Mills/Month)					
Max. Denting in Any Single Unplugged Tube (Tube Circum. Ave) (Mills)					

Denting (Con't)

Support Plate Levels	Max. Denting in Any Single Tube in Bundle Area (Tube Ave) (Mills) (1)					% of Tubes Affected By Denting in Bundle Area				
	a	b	c	d	e	a	b	c	d	e
1										
2										
3										
4										
5										
6										
7										
8										
9										
10										
11										
12										

TABLE KEY

NOTE: All percentages refer to the percent of the tubes within a given area of the tube bundle.

(1)

Area of the Tube Bundle	No. of Tubes Within the Area
a. Periphery of Bundle (wi/20rows for B&W; wi/10 rows for C.E. and W)	
b. Patch Plate (wi/4 rows)	
c. Missing Tube Lane (B&W only) (wi/5 rows)	
c. Flow Slot Areas (C.E. and W only) wi/10 rows)	
d. Wedge Regions (C.E. and W only) (wi/8 rows)	
e. Interior of Bundle (remainder of tubes)	

(2)

Allowable Limit for Wastage/Cavitation Erosion:

Allowable Limit For Denting:

(3)

1. Specifies area between the tube sheet and the first support plate

2. Specifies in the following locations: (list the additional locations)

   Wastage/Cavitation Erosion:

   Cracking:

(4)

Specify the date of the inspection for which results have been tabulated.

VIII. SIGNIFICANT STEAM GENERATOR ABNORMAL OPERATIONAL EVENTS

DATE	SUMMARY
	(Include event description; unscheduled ISI results, if performed; and subsequent remedial actions)

IX. CONDENSER INFORMATION

Condenser Material	Tube Leakage Date	Tube Leakage Rate (gpm)	Detectable Limit	Detection Method

X. RADIATION EXPOSURE HISTORY WITH RESPECT TO STEAM GENERATORS

Date	Exam Dosage (Man-Rem)	Repair Dosage (Man-Rem)	Comments

XI. DEGRADATION HISTORY FOR EACH TYPE OF DEGRADATION EXPERIENCED FOR TEN REPRESENTATIVE, UNPLUGGED TUBES FOR WHICH THE RESULTS OF TWO OR MORE ISI'S ARE AVAILABLE

If the results for ten tubes are not available, specify this information for all those tubes for which results are available.

(repeat the following information for each tube and degradation type)

Steam Generator No:
Tube Identification:
Type of Degradation: (specify denting, wastage, cavitation erosion, caustic stress corrosion cracking, or flow induced vibration cracking)

(repeat the following information chronologically for each ISI for which results are available)

ISI Date:
Amount of Degradation: (specify amount and units)
EFP Months of Operation Since Last ISI for Which Results are Given:

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

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cc: White Plains Public Library
100 Martine Avenue
White Plains, New York 10601

Leonard M. Trosten, Esquire
LeBoeuf, Lamb, Leiby & MacRae
1757 N Street, N.W.
Washington, D. C. 20036

Anthony Z. Roisman, Esquire
Sheldon, Harmon & Roisman
1025 15th Street, N.W.
Washington, D. C. 20005

Paul S. Shemin, Esquire
Assistant Attorney General
State of New York
Department of Law
Two World Trade Center
New York, New York 10047

Sarah Chasis, Esquire
Natural Resources Defense Council
122 East 42nd Street
New York, New York 10017

Director, Technical Development
Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Rear Admiral P. J. Early (IP-3)
Assistant Chief Engineer - Projects
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

Mr. P. W. Lyon
Manager - Nuclear Operations
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Mr. J. P. Bayne, Resident Manager
Indian Point 3 Nuclear Power Plant
P. O. Box 215
Buchanan, New York 10511

Dr. J. W. Blake
Manager - Environmental
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Docket file

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

January 25, 1978

50-247
~~50-286~~

All PWR Licensees (Except for Trojan, North Anna, Indian Point 3, Beaver Valley and St. Lucie 1)

Gentlemen:

In October of 1975, the NRC staff notified each licensee of an operating PWR facility of a potential safety problem concerning the design of the reactor pressure vessel support system. Those letters requested each licensee to review the design basis for the reactor vessel support system for each of its PWR facilities to determine whether certain transient loads, which were described in the enclosure to the letter, had been appropriately taken into account in the design. Furthermore, these letters indicated that, on the basis of the results of licensees' reviews, a reassessment of the reactor vessel support design for each operating PWR facility may be required.

Licensee responses to that request indicated that these postulated asymmetric loads have not been considered in the design basis for the reactor vessel support system, reactor internals including the fuel, steam generator supports, pump supports, emergency core cooling system (ECCS) lines, reactor coolant system piping, or control rod drives.

Subsequently in June 1976, the NRC staff informed each PWR licensee that a reassessment of the reactor vessel support system design for each of its facilities was required. While the emphasis of these letters was primarily focused on the need to reassess the vessel support design for transient differential pressures in the annular region between the reactor vessel and the cavity shield wall and across the core barrel, we indicated that our generic review may extend to other areas in the nuclear steam supply system (NSSS) and that further evaluation may be required.

For your information, Enclosure 1 is a summary of the background and current status of our review efforts related to this generic concern.

Dupe of ~~7905150612~~

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We have now determined that an assessment of the potential for damage to other NSSS component supports (e.g., steam generator and pump supports), the fuel assemblies, control rod drives, and ECCS piping attached to the reactor coolant system due to loadings associated with postulated coolant system piping breaks is required. Our request for additional information transmitted to you in June 1976 has been revised both to clarify our original request and to identify the extension of our concerns to other areas in the NSSS, as identified above. A copy of this revised request for additional information is provided as Enclosure 2.

The revised request for additional information identifies a requirement that your assessment of potential damage to the reactor vessel and other NSSS component supports, reactor vessel, fuel and internals, attached ECCS lines and the control rod drives should include consideration of breaks both inside and outside of the reactor pressure vessel cavity. This assessment should be made for postulated breaks in the reactor coolant piping system, (secondary systems are not to be included), including the following locations:

a) Reactor vessel hot and cold leg nozzle safe ends
b) Pump discharge nozzle
c) Crossover leg
d) Hot leg at the steam generator (B&W and CE plants only)

A number of licensees, have presented to the NRC staff alternate proposals, other than to conduct a detailed analyses, to resolve this concern. Based upon our review of these proposals, we have concluded that these alternative proposals do not establish an acceptable basis for long term operation without a detailed assessment of the risk resulting from these postulated transient loading conditions. We have, however, concluded that the low probability for occurrence of an event which could result in these loads establishes an adequate basis to justify continued operation for a short term period.

The NRC staff will consider an analysis that is applicable to more than one specific plant if it can be adequately demonstrated that such an analysis is either representative or bounding for each plant concerned.

Additional guidance regarding loading combinations (safe shutdown earthquake loads, loss of coolant accident loads), will be provided by about March 1, 1978, following the conclusion of staff investigations in this area.

Please respond within 30 days of receipt of this letter, indicating your intent to proceed with an evaluation of the overall asymmetric loss of coolant accident (LOCA) loads as described herein. In addition, please submit to us, within 90 days, your detailed schedule for providing the required evaluation. Your schedule should be consistent with our desire to resolve this problem within two years and should clearly state your intent to demonstrate the safety of long term continued operation.

We are transmitting information copies of this letter to the Westinghouse, Combustion Engineering and Babcock & Wilcox Companies. If you have any questions or want any clarification on this matter, please call your NRC Project Manager.

Copies of this letter are being sent to all addressees on the current service lists for each docket.

Sincerely,

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
1. Background and Current Status
2. Revised Request for Additional Information

cc w/enclosure:
See attached listing

January 25, 1978

ENCLOSURE 1

BACKGROUND AND CURRENT STATUS OF THE NRC STAFF REVIEW OF ASYMMETRIC LOCA LOADS FOR PWR FACILITIES

On May 7, 1975, the NRC was informed by Virginia Electric & Power Company that an asymmetric loading on the reactor vessel supports resulting from a postulated reactor coolant pipe rupture at a specific location (e.g., the vessel nozzle) had not been considered by Westinghouse or Stone & Webster in the original design of the reactor vessel support system for North Anna, Units 1 and 2. It had been identified that in the event of a postulated instantaneous, double-ended offset LOCA at the vessel nozzle, asymmetric loading could result from forces induced on the reactor internals by transient differential pressure across the core barrel and by forces on the vessel due to transient differential pressures in the reactor cavity. With the advent of more sophisticated computer codes and the accompanying more detailed analytical models, it became apparent that such differential pressures, although of short duration, could place a significant load on the reactor vessel supports and on other components, thereby possibly affecting their integrity. Although this potential safety concern was first identified during the review of the North Anna facilities, it has generic implications for all PWRs.

Upon closer examination of this situation, it was determined that postulated breaks in a reactor coolant pipe at vessel nozzles were not the only area of concern but rather that other pipe breaks in the reactor coolant system could cause internal and external transient loads to act upon the reactor vessel and other components. For the postulated pipe break in the cold leg, asymmetric pressure changes could take place in the annulus between the core barrel and the vessel. Decompression could occur on the side of the vessel annulus nearest the pipe break before the pressure on the opposite side of the vessel changes. This momentary differential pressure across the core barrel could induce lateral loads both on the core barrel and on the reactor vessel. Vertical loads could also be applied to the core internals and to the vessel due to the vertical flow resistance through the core and asymmetric axial decompression of the vessel. Simultaneously, for vessel nozzle breaks, the annulus between the reactor and biological shield wall could become asymmetrically pressurized resulting in a differential pressure across the vessel causing additional horizontal and vertical external loads on the vessel. In addition, the vessel could be loaded by the effects of initial tension release and blowdown thrust at the pipe break. These loads could occur simultaneously. For a reactor vessel outlet break, the same type of loadings could occur, but the internal loads would be predominantly vertical due to more rapid decompression of the upper plenum.

Although the NRC staff's original emphasis and concern were focused primarily on the integrity of the reactor vessel support system with respect to postulated breaks inside the reactor cavity (i.e., at a nozzle), it has since become apparent that significant asymmetric forces can also be generated by postulated pipe breaks outside the cavity and that the scope of the problem is not limited to the vessel support system itself. For such outside-cavity postulated breaks, the aforementioned concerns, such as the integrity of fuel assemblies and other structures, need to be examined.

In June 1976, the NRC requested all operating PWR licensees to evaluate the adequacy of the reactor system components and their supports at their facilities with respect to these newly-identified loads.

In response to our request, most licensees with Westinghouse plants proposed an augmented inservice inspection program (ISI) of the reactor vessel safe-end-to-end pipe welds in lieu of providing an evaluation of postulated piping failures. Licensees with Combustion Engineering plants submitted a probability study (prepared by Science Applications, Inc.) in support of their conclusion that a break at a particular location (vessel nozzle) has such a low probability of occurrence that no further analysis is necessary. A similar study has been recently submitted by Science Applications, Inc. (SAI) for B&W plants.

When the Westinghouse and CE owners group reports were received in September 1976, the NRC formed a special review task group to evaluate these alternative proposals. In addition, EG&G Idaho, Inc., was contracted to perform an independent review of the SAI probability study submitted for the CE owners group.

This review effort resulted in a substantial number of questions which previously have been provided to representatives of each group. Based on the nature of these questions and other factors to be discussed later in this report, we cannot accept these reports in their present form as a resolution for the asymmetric LOCA load generic issue. Based on our review, we have concluded that a sufficient data base does not presently exist within the nuclear industry to provide satisfactory answers to these information needs. Several long-term experimental programs would be required to provide much of this information. Although the probability study recently submitted by SAI for certain B&W owners does respond to some of the informal questions raised during our review of the SAI report prepared by CE plants, the more fundamental questions remain. Therefore, this conclusion also applies to the SAI topical report for B&W plants (SAI-050-77-PA).

A second - and equally important - reason for not accepting probability/ISI approaches as a solution at this point concerns our need and industry's need to gain a better understanding of the problem. We consider it essential that an understanding of the important breaks and associated consequences be known before applying any remedy - be it pipe restraints, probability, ISI, or some combination of these measures. Only in this way will we have a basis on which to judge the importance of the remedy with respect to what it is designed to prevent.

Although we have many questions on each of these topical reports, this does not mean that we view the probabilistic/ISI approach as completely without merit. In fact, the results of a probabilistic evaluation serves as the basis for continued operation and licensing of nuclear plants during this interim period while additional evaluations can be performed by vendors and licensees.

We believe that the justification for continued plant operation has as its basic foundation the fact that the event in question, i.e., a hypothetical double-ended instantaneous rupture of the main coolant pipe at a particular location, has a very low probability of occurrence.

The disruptive failure probability of a reactor vessel itself has been estimated to lie between 10^{-6} and 10^{-7} per reactor year - so low that it is not considered as a design basis event. The rupture probability of pipes is estimated to be higher. WASH-1400 used a median value of 10^{-4} for LOCA initiating ruptures per plant-year for all pipes sizes 6" and greater (with a lower and upper bound of 10^{-5} and 10^{-3}, respectively). We believe that considering the large size of the pipes in question (up to 50" O.D. and 4-1/8" thick), the lower bound is more appropriate since these pipes are more like vessels in size. In addition, the quality control of this piping is the best available and somewhat better than that of the piping used in the WASH-1400 study.

These factors, coupled with the facts that (1) the break of primary concern must be very large, (2) it must occur at a specific location, (3) the break must occur essentially instantaneously, and (4) these welds are currently subject to inservice inspection by volumetric and surface techniques in accordance with ASME Code Section XI, lead us to conclude that the probability of a pipe break resulting in substantial transient loads on the vessel support system or other structures is acceptably small such that continued reactor operation and continued licensing of facilities for operation can continue while this matter is being resolved.

In support of the above, the staff has developed a short-term interim criterion to determine if an acceptable level of safety exists for operating PWRs under conditions of a postulated pipe break. This interim criterion is based on a simplified probabilistic model that incorporates elastic fracture mechanics techniques to estimate the probability of a pipe break. Critical flaw size and subcritical flaw growth rates were determined assuming the presence of a surface flaw located in a circumferential weld of a thick-walled pipe, Determination of the critical flaw size was based on an estimated fracture toughness value of K_{IC} at a minimum temperature of 200 F and a uniform tensile stress equal to the consideration of various operating conditions producing elastically calculated stresses ranging in value from 1 to 3 times the material minimum yield strength.

Then, using the calculated critical flaw size, the subcritical growth rate, and an estimated probability distribution of an undetected flaw in thick-walled pipe welds, the upper bound probability of pipe break was estimated to be 10^{-4}. This value is also supported by a recent publication by Dr. S. H. Bush* which states that actual failure statistics confirm rates of 10^{-4} to 10^{-6} per reactor-year in large pipes, with higher rates as the pipe size decreases. Considering these analyses, we conclude that our conservative estimate on a pipe break in the primary coolant system is in the range of 10^{-4} to 10^{-6}. This estimated pipe break probability is considered acceptably low to justify short-term operation of nuclear power plants.

In view of all previous discussions concerning this issue, the NRC staff has concluded that an evaluation must be undertaken to assess the design adequacy of the reactor vessel supports and other affected structures and systems to withstand asymmetric LOCA loads, including an assessment of the effects of asymmetric loads produced by various pipe breaks both inside and outside the reactor cavity. On performing these evaluations the staff will permit the grouping of plants, where adequate justification for such grouping exists, in order to limit the number of plants to be analyzed. Alternatively, the staff will permit the analyzing of a "prototypical" plant, which is sufficiently representative of a group of plants, to provide the necessary information. Both of these concepts have been discussed with the Westinghouse and CE Owners Groups, and we believe that such approaches could save a significant amount of time and effort in obtaining results on which to base any needed corrective measures. The NRC staff is prepared to meet with PWR licensees to discuss such approaches, and has already done so. For example, we met with the Westinghouse owners group on October 19, 1977 for the purpose of discussing a generic solution for breaks outside the reactor cavity. It is expected that a similar meeting will be held in the near

*"Critical Factors in Blowdown Loads in the PWR Guillotine Nozzle Break (Volume 2 - the Asymmetric Load Problem)", ~~dated~~ June 6, 1977

future to address breaks located inside the cavity. This "phased" approach is acceptable to us, provided that it sheds light on and serves to expedite consideration of the more limiting inside-cavity breaks.

For your information, the NRC has a technical assistance contract with EG&G Idaho, Inc., to independently model representative Westinghouse, B&W, and CE plants for the purpose of assessing the loads on all major structures and components resulting from asymmetric LOCA loads. We believe that the results of this program which will include sensitivity studies, will provide significant confirmatory information related to this generic safety concern.

Although, as stated earlier, we believe that continued operation and licensing of facilities for the short-term is justified, we also believe that efforts to resolve this issue should proceed without delay, with the objective of both completing the necessary assessments and installing any necessary plant modifications within two years. In making this statement, we wish to make it clear that plant modifications, if indicated by licensee assessments, is the preferred approach. At the same time, we recognize that there may be cases wherein appropriate modifications may be judged to be unwarranted based on the consideration of overall risk. In such cases, and only in such cases, we will be prepared to give further consideration to alternate approaches, such as probability/ISI. We feel, however, that ISI techniques as they exist today could be considerably improved, and, to the extent that such improvements could have a direct bearing on this problem as well as an impact of nuclear safety in general, we would welcome their development.

ENCLOSURE 2

REVISED REQUEST FOR ADDITIONAL INFORMATION

Recent analyses have shown that certain reactor system components and their supports may be subjected to previously underestimated asymmetric loads under the conditions that result from the postulation of ruptures of the reactor coolant piping at various locations. It is therefore necessary to reassess the capability of these reactor system components to assure that the calculated dynamic asymmetric loads resulting from these postulated pipe ruptures will be within the bounds necessary to provide high assurance that the reactor can be brought safely to a cold shutdown condition. For the purpose of this request for additional information the reactor system components that require reassessment shall include:

a. Reactor Pressure Vessel
b. Fuel Assemblies, Including Grid Structures
c. Control Rod Drives
d. ECCS Piping that is Attached to the Primary Coolant Piping
e. Primary Coolant Piping
f. Reactor Vessel, Steam Generator and Pump Supports
g. Reactor Internals
h. Biological Shield Wall and Neutron Shield Tank (where applicable)
i. Steam Generator Compartment Wall

The following information should be included in your reassessment of the effects of postulated asymmetric LOCA loads on the above-mentioned reactor system components and the reactor cavity structure.

1. Provide arrangement drawings of the reactor vessel, the steam generator and pump support systems to show the geometry of all principal elements and materials of construction.

2. If a plant-specific analysis will not be submitted for your plant, provide supporting information to demonstrate that the generic plant analysis under consideration adequately bounds the postulated accidents at your facility. Include a comparison of the geometric, structural, mechanical and thermal hydraulic similarities between your facility and the case analyzed. Discuss the effects of any differences.

3. Consider postulated breaks at the reactor vessel hot and cold leg nozzle safe ends, pump discharge nozzle and crossover leg that result in the most severe loading conditions for the above-mentioned

systems.* Provide an assessment of the effects of asymmetric pressure differentials on these systems/components in combination with all external loadings including asymmetric cavity pressurization for both the reactor vessel and steam generator which might result from the required postulate. This assessment should consider:

a. limited displacement break areas where applicable
b. consideration of fluid-structure interaction
c. use of actual time-dependent forcing function
d. reactor support stiffness
e. break opening times.

4. If the results of the assessment required by 3 above indicate loads leading to inelastic action in these systems or displacement exceeding previous design limits provide an evaluation of the following:

   a. Inelastic behavior (including strain hardening) of the material used in the system design and the effect on the load transmitted to the backup structures to which these systems are attached.

5. For all analysis performed, include the method of analysis, the structural and hydraulic computer codes employed, drawings of the models employed and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.

6. Provide an estimate of the total amount of permanent deformation sustained by the fuel spacer grids. Include a description of the impact testing that was performed in support of your estimate. Address the effects of operating temperatures, secondary impacts, and irradiated material properties (strength and ductility) on the amount of predicted deformation. Demonstrate that the fuel will remain coolable for all predicted geometries.

7. Demonstrate that active components will perform their safety function when subjected to the postulated loads resulting from a pipe break in the reactor coolant system.

8. Demonstrate functionability of any essential piping where service level B limits are exceeded.

In order to review the methods employed to compute the asymmetrical pressure differences across the core support barrel during subcooled portion of the blowdown analysis, the following information is requested:

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*B&W and CE plant licensees should also consider breaks in the hot leg at the steam generator inlet.

ENCLOSURE 2

REVISED REQUEST FOR ADDITIONAL INFORMATION

Recent analyses have shown that certain reactor system components and their supports may be subjected to previously underestimated asymmetric loads under the conditions that result from the postulation of ruptures of the reactor coolant piping at various locations. It is therefore necessary to reassess the capability of these reactor system components to assure that the calculated dynamic asymmetric loads resulting from these postulated pipe ruptures will be within the bounds necessary to provide high assurance that the reactor can be brought safely to a cold shutdown condition. For the purpose of this request for additional information the reactor system components that require reassessment shall include:

a. Reactor Pressure Vessel
b. Fuel Assemblies, Including Grid Structures
c. Control Rod Drives
d. ECCS Piping that is Attached to the Primary Coolant Piping
e. Primary Coolant Piping
f. Reactor Vessel, Steam Generator and Pump Supports
g. Reactor Internals
h. Biological Shield Wall and Neutron Shield Tank (where applicable)
i. Steam Generator Compartment Wall

The following information should be included in your reassessment of the effects of postulated asymmetric LOCA loads on the above-mentioned reactor system components and the reactor cavity structure.

1. Provide arrangement drawings of the reactor vessel, the steam generator and pump support systems to show the geometry of all principal elements and materials of construction.

2. If a plant-specific analysis will not be submitted for your plant, provide supporting information to demonstrate that the generic plant analysis under consideration adequately bounds the postulated accidents at your facility. Include a comparison of the geometric, structural, mechanical and thermal hydraulic similarities between your facility and the case analyzed. Discuss the effects of any differences.

3. Consider postulated breaks at the reactor vessel hot and cold leg nozzle safe ends, pump discharge nozzle and crossover leg that result in the most severe loading conditions for the above-mentioned

systems.* Provide an assessment of the effects of asymmetric pressure differentials on these systems/components in combination with all external loadings including asymmetric cavity pressurization for both the reactor vessel and steam generator which might result from the required postulate. This assessment should consider:

a. limited displacement break areas where applicable
b. consideration of fluid-structure interaction
c. use of actual time-dependent forcing function
d. reactor support stiffness
e. break opening times.

4. If the results of the assessment required by 3 above indicate loads leading to inelastic action in these systems or displacement exceeding previous design limits provide an evaluation of the following:

   a. Inelastic behavior (including strain hardening) of the material used in the system design and the effect on the load transmitted to the backup structures to which these systems are attached.

5. For all analysis performed, include the method of analysis, the structural and hydraulic computer codes employed, drawings of the models employed and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.

6. Provide an estimate of the total amount of permanent deformation sustained by the fuel spacer grids. Include a description of the impact testing that was performed in support of your estimate. Address the effects of operating temperatures, secondary impacts, and irradiated material properties (strength and ductility) on the amount of predicted deformation. Demonstrate that the fuel will remain coolable for all predicted geometries.

7. Demonstrate that active components will perform their safety function when subjected to the postulated loads resulting from a pipe break in the reactor coolant system.

8. Demonstrate functionability of any essential piping where service level B limits are exceeded.

In order to review the methods employed to compute the asymmetrical pressure differences across the core support barrel during subcooled portion of the blowdown analysis, the following information is requested:

*B&W and CE plant licensees should also consider breaks in the hot leg at the steam generator inlet.

1. A complete description of the hydraulic code(s) used including the development of the equations being solved, the assumptions and simplifications used to solve the equations, the limitations resulting from these assumptions and simplifications and the numerical methods used to solve the final set of equations. Provide comparisons with experimental data, covering a wide range of scales, to demonstrate the applicability of the code and of the modeling procedures of the subcooled blowdown portion of the transient. In addition, discuss application of the code to the multi-dimensional aspects of the reactor geometry.

   If an approved vendor code is used to obtain the asymmetric pressure difference across the core support barrel, state the name and version of the code used and the date of the NRC acceptance of the code.

2. If the assessment of the asymmetric pressure difference across the core support barrel is made without the use of a hydraulic blowdown code, present the methodology used to evaluate the asymmetric loads and provide justification that this assessment provides a conservative estimate of the effects of the postulated LOCA.

A compartment multi-node, space-time pressure response analysis is necessary to determine the external forces and moments on components. Analyses should be performed to determine the pressure transient resulting from postulated hot leg and cold leg reactor coolant system pipe ruptures within the reactor cavity and any pipe penetrations. If applicable, similar analyses should be performed for steam generator compartments that may be subject to pressurization where significant component support loads may result. This information can be provided to encompass a group of similarly designed plants (generic approach) or a purely plant specific (custom plant) evaluation can be developed. In either case, the proposed method of evaluation and principal assumptions to be used in the analysis should be provided for review in advance of the final load assessment.

For generic evaluations, perform a survey of the plants to be included and identify the principle parameters which may vary from plant to plant. For instance, this should include blowdown rate and geometrical variations in principle dimensions, volumes, vent areas, and vent locations. A typical or lead plant should be selected to perform sensitivity and envelope calculations. These analyses should include:

(1) nodal model development for the configuration representing the most restrictive geometry; i.e., requiring the greatest nodalization;

(2) the most restrictive configuration regarding vent areas and obstructions to flow should be analyzed; and,

(3) sensitivity to code data input should be evaluated; e.g., loss coefficients, inertia terms, vent areas, nodal volumes, and any other input data where there may be variations from plant to plant or uncertainty for the given plant.

These studies should be directed at evaluating the maximum lateral and vertical force and moment time functions, recognizing that models may be different for lateral as opposed to vertical load definitions.

The following is the type of information needed for both generic and custom plant evaluations. Although this request was primarily developed for reactor cavity analyses it may be applied to other component subcompartments by general application.

(1) Provide and justify the pipe break type, area, and location for each analysis. Specify whether the pipe break was postulated for the evaluation of the compartment structural design, component supports design, or both.

(2) For each compartment, provide a table of blowdown mass flow rate and energy release rate as a function of time for the break which results in the maximum structural load, and for the break which was used for the component supports evaluation.

(3) Provide a schematic drawing showing the compartment nodalization for the determination of maximum structural loads, and for the component supports evaluation. Provide sufficiently detailed plan and section drawings for several views, including principal dimensions, showing the arrangement of the compartment structure, major components, piping, and other major obstructions and vent areas to permit verification of the subcompartment nodalization and vent locations.

(4) Provide a tabulation of the nodal net-free volumes and interconnecting flow path areas. For each flow path, provide an L/A (ft^{-1}) ratio, where L is the average distance the fluid flows in that flow path and A is the effective cross sectional area. Provide and justify values of vent loss coefficients and/or friction factors used to calculate flow between nodal volumes. When a loss coefficient consists of more than one component, identify each component, its value and the flow area at which the loss coefficient applies.

(5) Describe the nodalization sensitivity study performed to determine the minimum number of volume nodes required to conservatively predict the maximum pressure load acting on the compartment structure. The nodalization sensitivity study should include consideration of spatial pressure variation; e.g., pressure variation circumferentially, axially and radially within the compartment. The nodal model development studies should show that a spatially convergent differential pressure distribution has been obtained for the selected evaluation model.

Describe the justify the nodalization sensitivity study performed for the major component supports evaluated, if different from the structural analysis model, where transient forces and moments acting on the components are of concern. Where component loads are of primary interest, show the effect of noding variations on the transient forces and moments. Use this information to justify the nodal model selected for use in the component supports evaluation.

If the pressurization of subvolumes located in regions away from the break location is of concern for plant safety, show that the selection of parameters which affect the calculations have been conservatively evaluated. This is particularly true for pressurization of the volume beneath the reactor vessel. In this case, a model which predicts the highest pressurization below the vessel should be selected for the evaluation.

NOTE: It has been our experience that for the reactor cavity, three regions should be considered (i.e., nodalized) when developing a total model. These are:

(1) the volume around or in the vicinity of the break location out to a radius approximated by the adjacent nozzles, and including portions of the penetration volume for some plants;

(2) the volume or region covering the upper reactor cavity, primarily the RPV nozzles other than the break nozzle; and

(3) the region encompassing the lower reactor cavity and other portions of the reactor cavity not included in Items (1) and (2).

(6) Discuss the manner in which movable obstructions to vent flow (such as insulation, ducting, plugs, and seals) were treated. Provide analytical and experimental justification that vent areas will not be partially or completely plugged by displaced objects. Discuss how insulation for piping and components was considered in determining volumes and vent areas.

(7) Graphically show the pressure (psia) and differential pressure (psi) response as functions of time for a representative number of nodes to indicate the spatial pressure response. Discuss the basis for establishing the differential pressure on structures and components.

(8) For the compartment structural design pressure evaluation, provide the peak calculated differential pressure and time of peak pressure for each node. Discuss whether the design differential pressure is uniformly applied to the compartment structure or whether it is spatially varied. If the design differential pressure varies depending on the proximity of the pipe break location, discuss how the vent areas and flow coefficients were determined to assure that regions removed from the break location are conservatively designed, particularly for the reactor cavity as discussed above.

(9) Provide the peak and transient loading on the major components used to establish the adequacy of the support design. This should include the load forcing functions (e.g., $f_X(t)$, $f_y(t)$, $f_Z(t)$) and transient moments (e.g., $M_X(t)$, $M_y(t)$, $M_Z(t)$) as resolved about a specific, identified coordinate system. The centerline of the break nozzle is recommended as the X coordinate and the center line of the vessel as the Z axis. Provide the projected area used to calculate these loads and identify the location of the area projections on plan and section drawings in the selected coordinate system. This information should be presented in such a manner that confirmatory evaluations of the loads and moments can be made.

Consolidated Edison Company
of New York, Inc.

cc: White Plains Public Library
100 Martine Avenue
White Plains, New York 10601

Leonard M. Trosten, Esquire
LeBoeuf, Lamb, Leiby & MacRae
1757 N Street, N.W.
Washington, D. C. 20036

Anthony Z. Roisman, Esquire
Sheldon, Harmon & Roisman
1025 15th Street, N.W.
Washington, D. C. 20005

Paul S. Shemin, Esquire
Assistant Attorney General
State of New York
Department of Law
Two World Trade Center
New York, New York 10047

Sarah Chasis, Esquire
Natural Resources Defense Council
122 East 42nd Street
New York, New York 10017

Director, Technical Development
Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Rear Admiral P. J. Early (IP-3)
Assistant Chief Engineer - Projects
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

Mr. P. W. Lyon
Manager - Nuclear Operations
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Mr. J. P. Bayne, Resident Manager
Indian Point 3 Nuclear Power Plant
P. O. Box 215
Buchanan, New York 10511

Dr. J. W. Blake
Manager - Environmental
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

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

Docket file

February 2, 1978

50-247/286

All PWR Licensees (except for Trojan)

Gentlemen:

During the course of responding to the staff's review of an application for license amendment on the Trojan Nuclear Plant, the licensee informed the NRC that the reactor cavity annulus seal ring (used as a water seal during refueling operations, and not removed during normal operations) and associated biological shielding over the reactor vessel cavity could become missiles in the event of a loss of coolant accident (LOCA) pipe break inside the reactor vessel cavity. At the Trojan Nuclear Plant, these missiles could affect the ability of the control rods to shut down the reactor. From our preliminary evaluation of the information provided to the NRC staff by the licensee, the Portland General Electric Company and by Westinghouse, Combustion Engineering, Babcock & Wilcox and Bechtel in telephone discussions on January 25 and 26, 1978, it appears that this problem could occur in other PWR facilities such as yours and could potentially pose a threat to the health and safety of the public in the event of a LOCA.

Therefore, pursuant to 10 CFR 50.54(f) of the Commission's regulations, you are hereby requested to deliver to the Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, Washington, DC 20555, within 20 days of the date of this letter, i.e., February 22 , 1978, the following information: (a) a statement as to whether the cavity annulus seal ring in your facility is left in place during normal operation or if biological shielding is installed in the reactor cavity annulus and, if the answer to (a) is yes; (b) when you will determine whether the cavity annulus seal ring or biological shielding could become a missile in your facility, and (c) a description of what you plan to do, and when, if the problem is found at your facility and (d) justification for continued operations until the problem has been resolved, such justification to support why continued operation will not create undue risk to the health and safety of the public.

A copy of this letter is being provided to each licensee's current service list.

Sincerely,

Victor Stello, Jr.

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Dupe of ~~7910300697~~

mail