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POWER AUTHORITY OF THE STATE OF NEW YORK

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August 24, 1978
IPO-148

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. Albert Schwencer, Chief
Operating Reactors Branch No. 1

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286

Dear Mr. Schwencer:

By letters dated July 22, 1975 and June 9, 1976 the Nuclear Reactor Regulation Staff requested from Consolidated Edison Company of New York, Inc. various information concerning the Indian Point 3 Nuclear Power Plant reactor vessel supports. Partial responses to those requests were forwarded to the Staff by letters dated August 15, 1975, September 4, 1975, November 14, 1975 and July 9, 1976.

On June 15, 1977, Consolidated Edison submitted a Proprietary Class 2 Westinghouse Report WCAP-9117, "Analysis of Reactor Coolant System for Postulated Loss-of-Coolant Accident: Indian Point 3 Nuclear Power Plant" and in that way completed licensee responses to the July 22, 1975 and June 9, 1976 information requests.

Subsequently, in January 1978, Consolidated Edison received a telecopy which requested some additional information. The Power Authority of the State of New York submits herewith responses to NRC inquiries of January 1978 concerning the Indian Point 3 Nuclear Power Plant Reactor Vessel Support Analysis Program.

Very truly yours,

P. J. Early

Paul J. Early
Assistant Chief Engineer-Projects

Att.
cc: Hon. George V. Begany
White Plains Public Library

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ATTACHMENT 1

Question: Discuss the basis for concluding that the pump outlet nozzle is the most severe break outside the RPV cavity.

Response:

Previous to the analyses described in WCAP 9117, the reactor coolant system was analyzed for the postulated break locations described in WCAP-8172-A [(1)] (and the consideration of additional break locations in the cross-over leg) with the exception of the vessel nozzle breaks which were not considered. These were dynamic analyses which considered all the applicable transient blowdown loads and all piping systems and system supports were shown to have acceptable stress levels. The consideration of pipe ruptures in the cross-over leg, steam generator primary nozzles safe ends, and reactor coolant pump primary nozzle safe ends provides assurance that the structural integrity of the broken loop is maintained. The analyses described in WCAP-9117 considered the worst case break locations relative to the effect on the reactor vessel and unbroken reactor coolant loops. In this respect, the pipe ruptures which produce the largest vessel motion (or largest applied loads on the reactor vessel) will have the worst case effects on the reactor vessel and unbroken reactor coolant loops.

The loads applied to the reactor vessel during blowdown can be separated into three categories:

(1) reactor internal hydraulic loads due to depressurization waves travelling into the vessel shell and around the vessel internals.

(2) reactor coolant loop mechanical loads due to the release of the normal operating loads at the postulated break location.

(3) reactor cavity pressure loads (for breaks at the reactor vessel safe end locations).

Since loop mechanical loads are applicable for all guillotine breaks, the magnitude of the internal hydraulic loads for various break locations will determine the most severe break location outside the RPV cavity.

Reactor internal hydraulic loads are more severe for breaks in the cold leg or hot leg than for breaks in the cross-over leg due to the dissipation of the depressurization waves as they propagate through the steam generator or reactor coolant pump. In addition, due to the nature of wave propagations in the reactor vessel (see Section 3.3. of WCAP-9117) pipe ruptures in the cold leg generate larger internal hydraulic loads than breaks in the hot leg. The pump outlet nozzle break is the only cold leg guillotine rupture postulated outside the vessel cavity by WCAP-8172-A, and is the most severe break location.

[(1)] "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loops," WCAP-8172-A, January 1975.

Question: Demonstrate, for the ECCS piping attached to the unbroken loops, that the necessary flow rate is maintained when Appendix F limits are used. Discuss the basis for using the criteria of 50% of uniform ultimate strain.

Response:

The two worst case ECCS lines in the unbroken loops, the RHR line in loop 32 and the accumulator line in loop 33, were analyzed and did not demonstrate yielding. In a piping system, if stresses are held below yield, the cross-sectional flow area will be maintained.

The topic of 50% of uniform ultimate strain need not be discussed since all stresses in the primary system piping are below the faulted condition limits given in Appendix F of the ASME Code for inelastic system analysis and inelastic component analysis. In fact, the maximum stress is 33,184 psi in the cold leg of loop 32 or only 1.77 S_y (S_y=18,000 psi). The bending moment is of primary concern in the demonstration of piping flow rate maintenance. This is related to the tendency to form a plastic hinge leading to significant changes in cross-sectional flow area of the primary pipe. Tests have been performed by W on ten-inch stainless steel pipe (SA376, Type 304) similar to that used in the Westinghouse NSSS. These tests indicated that the ratio of a plastic hinge moment to the bending moment that causes initial yielding is approximately 3.2. In terms of yield stress, the maximum stress in the test pipe at the point of plastic hinge formation is approximatley 3.2 S_y. Also from the tests it was observed that, at the point of initial plastic hinge formation, the angle of rotation of the pipe reached a maximum of less than 20 degrees. Even under this loading conditon, there was no discernible distortion of the cross section of the pipe. The tests were run without internal pressure, which is conservative since internal pressure will increase the stability against cross section collapse. Considering that the yielding in the primary piping was predicted only at localized areas, and that the magnitude of the stress (1.77 S_y) is much lower than the stress that has been shown by tests to cause a plastic hinge (3.2S_y), significant reduction in the primary pipe cross-sectional area will not occur.

Summarily, the loads in the ECCS piping andprimary system piping will not induce significant changes in cross-sectional flow area and the flow rate for core cooling following a LOCA will be maintained.

Question: Provide the loading combination considered in this analysis. Provide a discussion on the basis for not combining seismic and LOCA loads.

Response:

In June of 1976 a letter from the US NRC Operating Reactors Branch requested a reassessment of the reactor vesel supports for postulated loss-of-coolant accidents (LOCA). The letter noted several items to be included in the evaluations. No combination of seismic and LOCA responses was requested. After the utility contracted Westinghouse to perform the analyses, a report was prepared and submitted to the OperatingReactors Branch in June of 1976 entitled "Criteria for Analysis of Reactor Vessel Supports for Indian Point Unit 3". A meeting was held to discuss the proposed critera; no emphasis was placed upon the methods of load combinations. Section 2.5 of the referenced document states that the LOCA responses would not be combined with the seismic responses due to the neglibly small probability of a simultaneous earthquake and pipe rupture occurrence. (The original design and evaluation of Indian Point Unit 3 considered the simultaneous occurrence of a seismic event and pipe rupture and demonstrated plant safety for this response combination.) The seismic response of concern is that due to a safe shutdown earthquake (SSE). The occurrence of either an SSE or LOCA is a remote event. The definition of an SSE as the maximum credible earthquake which can be predicted at the plant site assures that the probability of an SSE occurrence is low. The occurrence of a pipe rupture is also a low probability event due to the stringent design criteria, the use of material which is highly resistent to fracture, and the comprehensive preservice and inservice inspection techniques employed. The unlikelihood of the events and the capability of the system to withstand the events assure that there is a high degree of structural integrity in the Indian Point primary coolant system. The combination of SSE and LOCA responses assumes that an SSE may induce a LOCA. Reference (2) demonstrated that for a typical high seismic plant, the probability of a LOCA resulting from an SSE is very low. These studies included consideration of undetected flaws in the pipe and concluded that, even with flaws much larger than those realistically expected in nuclear power plant piping, and in the worst location, the conditional probability of a LOCA resulting from large earthquake motions is on the order of 10^{-6} per SSE. It should be emphasized that this is a conditional probability and, therefore, does not include the probability of occurrence of the limiting earthquake itself, nor of the existence of a flaw. Based upon these relationships, postulation of an SSE induced LOCA does not constitute a reasonable evaluation basis. The combination of SSE and LOCA responses in the original design of nuclear power plants adds margin to the plants' design. This combination does not imply that the events are expected to occur simultaneously, but the margin to safety is increased for events which are more likely to occur.

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(2) "Integrity of Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events,"Witt, Bamford, Esselman, WCAP 9283, March 1978.

For plants which have passed the construction phase, like Indian Point Unit 3, the intent of the evaluations for events like those postulated in WCAP-9117 is to obtain a realistic assessment of the plant safety for the postulated events. It is not realistic to assume the simultaneous occurrence of an SSE and LOCA and, therefore, the combination was not considered.

Question: Discuss the potential for steam generator support failure due to pressurization of the S. G. cavity if an outside break occurs at the S.G. If the S.G. support fails, what is the consequence of a second break, in the primary coolant system, on containment integrity.

RESPONSE:

The region immediately surrounding the steam generator is not conducive to asymmetric pressurization because of the openness of the design. See Appendix C of WCAP-9117 for plant general arrangment drawings. Drawing A202078 reveals the openness of the loop compartments. There are no secondary shield walls surrounding the steam generators and thus no significant asymmetric loads would be generated for primary coolant system pipe rupture. The steamline runs above the biological shield wall in an open area and, consequently, a rupture in the vertical drop would cause no significant asymmetric pressurization. There would be little or no vertical force exerted on the steam generator as a result of the steam line rupturing at the top of the steam generator since the steam generators are not enclosed at the top. Thus, no significant loads are exerted on the steam generator as a result of pipe ruptures. The detailed analyses of the primary coolant system for various postulated break locations have verified the structural integrity of the steam generator supports.

Question: Page 3-12, Figure 3-7. Why isn't the vertical force initially (at time = 0 sec.) equal to zero?

RESPONSE:

Figures 3-7 and 3-10 include the effect of loads applied to the reactor vessel safe ends due to thermal, pressure, and deadweight effects from the attached piping. These loads are determined from normal operation condition analyses of the RC system and a downward vertical force of approximately 1.1 x 10^6 pounds is due to this effect. In addition, there is a normal operating condition pressure differential of approximately 27 psi between the upper head and lower dome of the reactor vessel which leads to a load on the vessel shell of approximately 700,000 pounds. These two effects combine to produce the 1.8 x 10^6 pounds applied to the vessel shell at the beginning of the blowdown transient in Figures 3-7 and 3-10.

Question: Page 3-77, paragraph 6. Are the cable elements capable of plastic action?

RESPONSE:

The cable elements are not capable of plastic deformation. The tie rods which the cable elements represent experience loads only in the elastic range.

Question: Page 4-2, paragraph 4-8. Were the vertical and rotational displacements considered in the static evaluation of the reactor coolant loop piping along with the horizontal displacement? Was the dynamic load factor applied to these displacements also?

RESPONSE:

The vertical and rotational displacement of the reactor vessel were not considered in the static evaluation of the reactor coolant loop. The analyses did consider the horizontal displacement with a dynamic load factor (DLF) of two. The conservatism of using only the peak horizontal displacement with a DLF of 2 is discussed in Appendix D of WCAP-9117.

Question: Page 3-77, paragraph 7. Are the stiffness matrices for loops 32 and 33 linear or nonlinear?

RESPONSE:

The stiffness matrices for loops 32 and 33 are nonlinear. The nonlinearity considered is due to the plastic shell-type local deformation of the primary piping at the pipe restraint locations.

Question: Page 3-81, paragraph 2. How do the load-deflection curves of figures 3-33 and 3-34 unload? What effect could unloading have on RPV and internal response?

RESPONSE:

In the reactor pressure vessel dynamic analysis, the reactor coolant loops are represented such that unloading of the loop stiffness matrix occurs plastically, i.e. after the elastic yield point is passed, unloading occurs parallel to the elastic portion of the stiffness-deflection curve. This is considered an accurate representation of the actual unloading phenomena and thus the effect upon the vessel and internals response is realistically determined. The effect of different (unrealistic) unloading representations should have a negligible effect upon the evaluations presented in WCAP-9117, as discussed in the response to the next question.

Question: Page 3-82, paragraph 2. Was unloading considered in the reactor shoe test? If so, how do they unload? What is the effect of the unloading on RPV and internals response (figure 3-38)?

RESPONSE:

In the testing of the reactor vessel shoe, unloading was not considered. It is reasonable to assume that unloading would occur parallel to the elastic portion of the load-deflection relationship. This type of unloading was employed in the reactor pressure vessel dynamic analysis. Since the peak vessel displacement and internals horizontal response generally occur during the first excursion of the vessel from its normal operating position, the peak response would remain the same if different unloading phenomena were considered. For example, the peak horizontal displacement of the vessel due to either vessel nozzle rupture occurs before unloading of the vessel horizontal restraint occurs (see figures 3-42 and 3-47). The transient response after this peak response would be slightly changed if different unloading characteristics were assumed, but since the loop piping and component analyses were performed statically with the peak displacement (with a DLF of 2) due to the vessel outlet nozzle rupture, the conclusions of this analysis would not change if different unloading characteristics were assumed. With respect to the reactor vessel internals and core evaluation, all peak responses occurred immediately after the first excursion of the vessel from its initial position and thus the unloading of the vessel restraint will not affect the peak responses of these components.

The unloading assumed in the reactor vessel dynamic analysis is considered a realistic representation of the unloading of the vessel restraints but, even if alternate unloading were assumed, the results of WCAP 9117 would not change.

Question: Page 4-26, paragraph 2. Discuss why modeling the pipe annulus opening would cause more of the load to be resisted in the vertical direction.

RESPONSE:

The evaluation of effects of pipe break loads at the Reactor Vessel inlet nozzles consisted of two operations. First a finite element analysis, which did not include the primary coolant pipe openings, was used to evaluate the membrane and bending loads. It was found that most of the load was carried in the hoop direction rather than the vertical direction. Hand calculations were then performed assuming all pressure loads were carried in the vertical direction by concrete beams between the primary coolant pipe openings. If the finite element model had included the pipe openings, the actual distribution of load would have been determined based on the relative stiffness of the hoop and vertical directions. Since the openings interrupt the load path in the hoop direction and reduce the stiffness, a greater portion of the load would be carried in the vertical direction. This would result in a reduction of the load in the hoop direction and an increase of the load in the vertical direction from that obtained in the model without the pipe opening. Therefore, the resulting hoop loads from the analysis are conservative.

Question: Provide a discussion of vent areas which your analysis takes credit for, which are normally, fully or partially blocked during normal operation and are expected to blow out. Include minimum pressure necessary to blow out the blocking structure and discuss the affect of the missile. Examples of these structures are seal rings, vent plugs, etc. (as applicable).

RESPONSE:

The assumptions made in the Reactor cavity pressure analysis are described on pages 3-17 and 3-24 of WCAP 9117. Credit was taken for venting through the inspection ports once the plugs have been expelled. The seal table is elevated above the supports and outside the surrounding concrete. Therefore, credit for venting out of the lower reactor cavity thru the instrumentation tunnel can be taken without blowing out any structures. The refueling cavity seal ring will not be in place during normal operation and consequently, credit for venting out of the top of the reactor vessel annulus was taken.

50-286

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

August 9, 1978

ALL PWR LICENSEES

Gentlemen:

The Division of Operating Reactors, Office of Nuclear Reactor Regulation, has organized a two-day PWR Steam Generator Conference to be held at the Holiday Inn in Bethesda, Maryland on September 7 and 8, 1978. The purpose of the conference is to provide an opportunity for industry, government, national laboratory, foreign, and possible public representatives to present and discuss operating experience relevant to steam generators and to exchange ideas for integrating design, inspection and operating procedures to ensure more reliable, safe operation of steam generators at nuclear power facilities.

Attached for your use is a Notice of the Conference and a tentative agenda.

Please notify Dr. B. D. Liaw, Division of Operating Reactors, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, telephone (301) 492-8060 of your intent regarding attendance at the conference by August 25, 1978.

Darrell Eisenhut

D. G. Eisenhut, Assistant Director
for Systems and Projects
Division of Operating Reactors

Enclosures:
1. Notice of Conference
2. Tentative Agenda

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ENCLOSURE NO. 1

CONFERENCE NOTICE

The Nuclear Regulatory Commission will sponsor a two-day Pressurized Water Reactor Steam Generator Workshop at the Holiday Inn in Bethesda, Maryland on September 7 and 8, 1978. The purpose of the workshop is to provide an opportunity for industry, government, national laboratory and foreign organizations, and possibly, public representatives to present and discuss operating experience relevant to steam generator tube degradation and to exchange ideas for integrating design, inspection and operating procedures to ensure safe operation of steam generators at nuclear power facilities. The workshop will be comprised of presentations by invited speakers followed by a panel discussion.

Requests for additional information, including requests to participate, should be addressed to Dr. B. D. Liaw, Division of Operating Reactors, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555. Telephone (301) 492-8060.

A tentative agenda of the workshop is attached.

ENCLOSURE NO. 2

PRESSURIZED WATER REACTOR STEAM GENERATOR WORKSHOP

DIVISION OF OPERATING REACTORS

OFFICE OF NUCLEAR REACTOR REGULATION

U. S. NUCLEAR REGULATORY COMMISSION

WASHINGTON, D. C. 20555

General Chairman: Darrell G. Eisenhut, Assistant Director for Systems and Projects
Division of Operating Reactors

September 7, 1978

8:00 a.m. - Registration

INTRODUCTORY SESSION: D. G. Eisenhut

9:00 a.m. - Opening and Welcome Remarks (V. Stello)

9:10 a.m. - Licensing Bases for Continued Operation of PWR Steam Generators (D. G. Eisenhut)

9:30 a.m. - NRC Confirmatory Research Programs (C. Z. Serpan)

- Coffee Break -

GENERAL SESSION: L. C. Shao

10:00 a.m. - Westinghouse Steam Generator Operating Experiences (Representative - Westinghouse Electric Corporation)

10:30 a.m. - Combustion Engineering Steam Generator Operating Experiences (Representative - Combustion Engineering)

11:00 a.m. - B&W Steam Generator Operating Experiences (Representative - Babcock & Wilcox, Inc.)

- Lunch Break -

TECHNICAL SESSION I: J. P. Knight

1:30 p.m. - Eddy Current Inspection Method Evaluation
(Representative - Battelle Columbus)

2:00 p.m. - Advanced ECT Probe Development
(Representative - ZETEC, Inc.)

2:30 p.m. - PNL Steam Generator Tube Integrity Program
(Representative - Pacific Northwest Laboratory)

- Coffee Break -

3:30 p.m. - BNL Stress Corrosion Tests
(Representative - Brookhaven National Laboratories)

4:00 p.m. - DOE Chemical Cleaning Program
(Representative - U. S. Department of Energy)

September 8, 1978

TECHNICAL SESSION II: B. D. Liaw

9:00 a.m. - Model Boiler Test for Reproducing Tube Denting
(Representative - Combustion Engineering, Inc.)

9:30 a.m. - Improved Westinghouse Steam Generator Design to Avoid Various Forms of Tube Degradation
(Representative - Westinghouse Electric Corporation)

10:30 a.m. - Experience with Condenser Failures, Retubing and Consequence
(Representative - Westinghouse Electric Corporation)

11:00 a.m. - Turkey Point Steam Generator Replacement Program
(Representative - Bechtel Power Corporation)

- Lunch Break -

1:30 p.m. - PANEL DISCUSSIONS: D. G. Eisenhut

Panel Members: J. R. Weeks, L. Frank, J. Muscara, J. Scinto, F. Almeter, B. D. Liaw, and various industry representatives

- Need for Secondary Water Chemistry Control
- Steam Generator Tube Denting, Support Plate Cracking and Deformation
- Regulation and Regulatory Guide Interpretations - Tube Plugging Criteria, ISI Requirements
- Development ECT Inspection Techniques
- Additional Research Programs

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