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In the Matter of Consolidated Edison Company of New York, Inc.
Indian Point Nuclear Generating Unit No. 2
Docket No. 50-247

Dear Mr. Roisman:

Transmitted herewith are the responses of the AEC regulatory staff to the ECCS questions you submitted on September 16, 1971.

Sincerely,

Myron Karman

Myron Karman
Counsel for AEC Regulatory Staff

Enclosure:
As stated

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hearing

QUESTION: Page 3

What are all of the changes that have been made in the analysis of the ECCS to correct the erroneous assumption regarding uniform and instantaneous mixing? For instance, have rod quench tests for shattering and rod swelling and bursting tests been redone to reflect variable temperatures in the coolant? How do the non-uniform and non-instaneous mixing affect the predictions on core internal remaining intact during blowdown and post blowdown pressures?

ANSWER: The thermal hydraulic computer programs currently used for the analysis of postulated loss-of-coolant accidents (LOCA) assume thermodynamic equilibrium (uniform and instantaneous mixing of water and steam). This assumption has been shown to be acceptable by semi-scale and other blowdown experiments (such as the Containment Systems Experiment) during the subcooled and saturated portions of the simulated blowdown transients. However, the 845-851 semi-scale test series did indicate that uniform and instantaneous mixing of the accumulator water in the lower plenum might not have occurred during the ECC injection phase of the transient. Since the assumption of thermodynamic equilibrium was not verified for the injection phase, the prediction by the codes concerning the fate of this fluid cannot be relied upon. This condition is conservatively accounted for in the AEC's Interim Policy Statement wherein it is assumed that all accumulator water injected up to the end of the primary system blowdown transient is assumed to be lost from the reactor system.

The assumption of thermodynamic equilibrium is not significant with respect to the various tests performed concerning the fuel cladding integrity. These tests demonstrated that cladding integrity was primarily a function of cladding temperature, and not coolant temperature. Since the Interim Policy Statement provides that a conservative calculation of peak cladding temperatures be performed, and clad integrity tests have been performed over the temperature range to establish the failure thresholds, further testing because of potential variable coolant temperatures is not necessary.

The most significant forces on the reactor internal structures for a postulated LOCA occur during the subcooled portion of the blowdown transient; i.e., within the first second following the postulated primary system failure. As discussed above, the assumption of thermodynamic equilibrium is acceptable up to the time of accumulator injection. During the period of accumulator injection, the forces on the internal structures are insignificant and would remain so considering non-equilibrium effects.

QUESTION: Pages 3-4

Which actual test results are relied upon by the Staff in its safety evaluation and in which is there not an exact scaling of parameters affecting system performance?

ANSWER: The staff considers tests which provide a better understanding of LOCA phenomenon relevent to its safety evaluation of a nuclear power plant.

The recent semi-scale tests (series 845-851) provided additional information related to the decompression and mixing processes ocurring during blowdown. The recently completed FLECHT series of heat transfer tests represented new information used by the Staff in its safety evalution of IPP-2. However, in all of these tests, an exact scaling of parameters was not a test requirement. Instead, a range of parameters was tested to investigate the behavior of LOCA phenomena for a variety of conditions.

QUESTION: Page 4

What are all of the inadequacies of the currently used calculational techniques to predict accumulator water behavior during blowdown and how have these and similar inadequacies been eliminated in predicting all of the post-blowdown behavior of the accumulators?

ANSWER: The function of the accumulator system following a postulated LOCA is to refill the reactor vessel quickly and terminate the fuel cladding temperature transient. The major uncertainties in these calculations are:

(1) the delivery of accumulator water during primary system blowdown and

(2) the reflood rate of the reactor core to provide cooling to the cladding hot spot.

These uncertainties are conservatively treated in the AEC's Interim Policy Statement as follows:

All accumulator water injected up to the end of blowdown is assumed to be lost from the primary system. When the fluid flow out the postulated break has ceased because of the increasing containment back pressure, a mechanism for further loss of accumulator water no longer exists and the remaining fluid fills the lower plenum and downcomer annulus.

Following the refill of the lower plenum up to the bottom of the core, the core reflood rate (inches/sec) is limited by the steam generated (steam binding) as the lower portions of the fuel rods are cooled. As the ability to vent this steam through various paths is increased, the core reflood rate is increased and the temperature transient is terminated sooner. The steam can be vented through two paths: (1) through the broken loop and out the break, and (2) through the intact loops to the downcomer annulus and then out the break. It is possible for the injection of accumulator water to block a portion of the vent area through the intact loops thereby reducing the ability to vent the steam. Since the degree of potential plugging is unknown, it is conservatively assumed that all intact loops are completely plugged (no steam relief) up to the end of accumulator water injection. The above conservatisms lengthen the time to get water to the core and reduce the core reflooding rate thereby increasing the calculated peak cladding temperatures.

QUESTION: Page 7

Is the 34 second lag for full rated flow of the pumping system based upon the most conservative assumption used in prior analysis, i.e., have the worst credible diesel and pump failures been considered? Is the 34 seconds the same assumption used by the Applicant in its most recent analysis of the post-LOCA conditions?

ANSWER: The 34 second delay to full rated flow of the ECCS pumps reported in the Safety Evaluation is broken down as follows: (FSAR p6.2-40)

To initiate safety injection signal including instrument lag	1 sec.
To start two diesel generators	19 sec.
To start two safety injection pumps	8 sec.
To start one residual heat removal pump	6 sec.
Total	34 sec.

The diesel generators at Indian Point Plant 2 are designed to start and come up to speed within ten seconds after initiation. Recent general experience indicates that diesel generators at nuclear plants can be started within approximately 10 seconds. Therefore it can reasonably be expected that the delay time to full flow would be about 25 seconds. The applicant has used a 25 second delay time in his calculation of ECCS performance following a LOCA. Since ECCS performance during the early stages of core recovery is dependent on accumulator injection, a difference in the delay time for pumped flow of 9 seconds has a negligible effect on the maximum clad temperature reached in the transient.

The above delay times includes the assumption of single failure, that is, full rated flow from two of three safety injection pumps and one of two residual heat removal pumps powered by two of three diesel generators would normally be available within 25 seconds following the safety injection signal.

QUESTION: Page 8

In what way will the operation of the reactor be affected by modifications in the nuclear hot channel factors?

ANSWER: No change in operation of the reactor will result from modification (reduction) of the nuclear hot channel factors. The large body of startup and operating data from the R. E. Ginna, Point Beach, H. B. Robinson, and European reactors using zircaloy clad fuel show that operating peaking factors can be maintained well below the conservative values used in the design of early powerplants, for which data and operating experience were lacking.

The hot channel factors initially used for the Indian Point II safety analysis were $F_q^N = 3.12$ (heat flux nuclear hot channel factor), and $F_{\Delta H}^N = 1.75$ (enthalpy rise hot channel factor). At the design value of $F_{\Delta H}^N$ of 1.75, a value of 1.78 would be permissible for F_z^N, the axial peaking factor ($F_q^N = F_{\Delta H}^N \times F_z^N$). With the reduced values of $F_q^N = 2.90$, and $F_{\Delta H}^N = 1.66$, the appropriate reduced F_z^N is 1.75. Technical specification changes will be made incorporating the lower values.

The original design value of 1.75 for $F_{\Delta H}^N$ included a 10% measurement uncertainty factor; the reduction of $F_{\Delta H}^N$ to 1.66 reflects a 5% reduction in the measurement uncertainty. We believe that $F_{\Delta H}^N$ can be determined by measurement within 5%, as described in WCAP-7308-L. Further, the operating data and experience show that the $F_{\Delta H}^N$ will normally be substantially below the value of 1.66. Startup and periodic operating measurements will verify achievement and maintenance of $F_{\Delta H}^{N}$ below 1.66. A value of 1.66 (the measured value would be 5% less) might only be approached in an abnormal situation such as a misaligned control rod, for which the Technical Specifications require confirmatory measurements for continued full power operation.

Operating data and experience also show that F_z^N can be maintained well below the original Indian Point II design value of 1.78. We have accepted design values between 1.67 and 1.73 for all of the Westinghouse designed reactors following Ginna, Robinson, and Indian Point II. All have 12 foot long cores and no design difference which would affect the basic axial power shape. Further, there is continuous monitoring of the axial power shape by the ex-core detectors and a well established correlation between their readings and F_z^N which can permit control of F_z^N to values as low as 1.55. No problem is therefore foreseen in maintaining F_z^N at or below the reduced value of 1.75.

QUESTION: Page 8

To what extent have the Idaho tests (845-851) been taken into account in predicting when blowdown will be completed, how much accumulator water actually is lost during blowdown, the path of steam during and after blowdown.

ANSWER: For the evaluation of a postulated LOCA for the Indian Point 2 plant, the Idaho tests (Semiscale Test Series 845-851) cannot be used directly to establish when blowdown will be completed. The Idaho tests were performed with the ruptured pipe discharging to atmosphere. For this condition, blowdown was not observed to end until the semiscale system reached atmospheric pressure. (For Test No. 848, this was not observed to occur until 90 seconds after the pipe rupture was initiated.) For Indian Point 2, the termination of blowdown is calculated in accordance with Paragraph No. 5 of the Westinghouse Evaluation Model. (Part 3, Appendix A of AEC Interim Policy Statement) The specific time for the end of blowdown is a function of the postulated size of the rupture. For the postulated double ended rupture of a cold leg pipe, the calculated end of blowdown occurs about 15 seconds after the break. In accordance with Paragraph 5 of the Westinghouse Evaluation Model, all of the water injected during blowdown is assumed to be lost from the system. The amount of water injected during this period is 565 cubic feet of water or 26 percent of the initial inventory of 3 accumulator tanks. This arbitrary assumption of loss of accumulator water is in addition to the assumed complete loss of the fourth accumulator connected to the cold leg pipe in which the rupture occurs. During blowdown in the Semiscale tests, a large fraction of the discharging fluid passed through the core. This core flow and the significant heat transfer which resulted from it was not considered in the safety evaluation of IPP-2. Because of the extended time for blowdown of Semiscale and the small amount of ECC water remaining in the vessel, post blowdown steam flow was negligible.

QUESTION: <u>Pages 8-9</u>

What other changes if any have been made in the ECCS performance calculations? Please explain all of these changes in greater detail.

ANSWER: All of the changes to the analysis assumptions from those previously used in the ECCS performance calculations have been listed on pages 7 and 8 of the Staff's Supplement 3 of its Evaluation of Indian Point 2 plant. These changes consisted of:

1) A 5% reduction in the assumed nuclear peaking factor. This change resulted in a change in the term normally added to the maximum heat flux calculated for the steady state, transient or accident condition to account for uncertainties in the measurement. The uncertainty term was reduced from 10% to 5%. Westinghouse has performed measurements on operating plants to determine the uncertainties associated with flux mapping measurements using a moveable incore detector system. These studies were combined with critical experiments to assess uncertainties associated with the measurement of various factors used to determine the peak nuclear heat flux factor. Based on this work it was determined that the probability of not exceeding an uncertainty of 4.58% in the peak nuclear heat flux factor is 95% at a 95% confidence level. Accordingly, the Staff concluded that a reduction in the error allowance to 5% was acceptable.

2) A change in the model for calculating the resistance of the reactor upper core support plate. In previous analysis, the hydraulic resistance and flow area between the upper head and the upper plenum and the upper head and the downcomer were assumed to be equal. A more realistic representation of the hydraulic resistance and the flow area has now been applied to these flow paths. A more detailed discussion is presented in Section 6.5 of the Westinghouse Report "Emergency Core Cooling Performance," dated June 1, 1971.

3) A 20% increase in the decay heat with a decrease in the heat deposition in the hot rod from 97.4% at steady state to 95% for the loss of coolant accident. The Commission's Interim Policy Statement (Appendix A) required that all evaluation models use the decay heat curve described in the proposed ANS Standard, with an added 20% allowance for uncertainty. The reduction in heat production for the hot rod results from consideration of the change in the heat generation profile of the hot rod which would occur when voiding occurred in the core during the early stages of the accident. Additional details of residual decay heat is presented in Appendix B to Additional Testimony of Applicant Concerning Emergency Core Cooling System Performance, Dated July 13, 1971.

The decay heat rate used by Westinghouse in its earlier analysis of the Indian Point 2 plant had an error resulting in a greater contribution to the residual decay heat from the heavy element (U-239 and Np-239) decay of neutron capture products. Correcting this error and following the Commission's Interim Policy Statement resulted in a decay heat slightly lower than the earlier prediction.

4) A 20% reduction in core flow when applied to hot channel calculations. The previous analysis utilized the calculated mid-plane core flow from the SATAN code as input to the hot rod heat-up code LOCTA-R2. In the reanalysis, the flow applied to the hot channel was reduced 20% to provide an additional conservatism to the analysis.

5) The assumption that the time to departure from nucleate boiling is equal to 0.1 seconds. The previous analysis was based on the calculated time to departure from nucleate boiling (DNB) of 0.5 seconds. Paragraph No. 4 of the Westinghouse Evaluation Model requires that heat transfer from a nucleate boiling regime be limited to 0.1 seconds after the break. This short time period was chosen because the predicted local fluid conditions were outside the range

of experiments from which the DNB correlation was formulated.

6) A revision to the transition boiling correlation. The heat transfer correlation for the transition boiling regime was modified to permit increased heat transfer for conditions in which the wall superheat temperature did not exceed 250°F based on heat transfer experiments performed by Westinghouse. A discussion of this modification to the correlation is presented in Section 6.2 of the Westinghouse Report, "Emergency Core Cooling Performance," dated June 1, 1971.

QUESTION: Page 9

Explain the offsetting effect of the changes and the basis for making the changes, - i.e. test results, etc. Does the 1550°F temperature occur when the most conservative assumptions (as used in earlier staff and Applicant analysis) are used?

ANSWER: Changes 1, 2, 3, and 6 discussed in the previous answer result in lower peak clad temperatures. Changes 4 and 5 result in an increased clad temperature. A discussion of the bases for these changes is presented in the answer to the previous question.

The peak clad temperature of 1550°F occurring at the end of blowdown was obtained from the new calculation which included the six changes to the blowdown portion of the analysis. The 1550°F temperature thus represents a calculation which includes a combination of additional conservatism in certain parts of the calculation and a more realistic representation in other areas.

QUESTION: Page 10

Under the new analysis, what are the rod temperatures at each second following the LOCA until reflooding begins and to what extent do these differ from the rod temperature behavior as originally predicted. To what extent have these rod temperature variations been taken into account in determining flow blockage from rod swelling and bursting and core disassembly such as rod shattering?

ANSWER: The table below lists the figures which include curves of peak clad temperature for the spectrum of breaks.

Break Size	Reevaluated Basis	Original Basis
Double Ended	Figure 10[1]	Figure 4[3]
0.8 Double Ended	Figure 11[1]	
4.5 ft^2	Figure 12[1]	
3.0 ft^2	Figure 13[1]	
0.5 ft^2	Figure 2[2]	Figure 14.14[4]

Cladding temperature is not considered to have an adverse effect on core geometry so long as the peak clad temperatures are less than 2300°F. Even if the hot spot is at 2300°F and some ballooning and rupture has occurred, less than 10% of the core cladding is above 2000°F. No adverse effect of flow blockage was observed in the PWR FLECHT tests, in fact, ECC heat transfer was enhanced by local flow blockage. Also in BWR

1) Provided in Additional Testimony of Applicant "Emergency Core Cooling System Performance," dated July 13, 1971.
2) Provided in letter to P. A. Morris, dated August 16, 1971.
3) Provided in Appendix 14B as Supplement 12 to the IPP No. 2 FSAR (Docket No. 50-247).
4) Provided in Supplement 13 to the IPP No. 2 FSAR (Docket No. 50-247).

FLECHT tests although local clad fragmentation occurred, no degradation in heat transfer was detected. On the basis of the information available we have concluded that so long as cladding temperatures are kept below 2300°F at the hot spot in present reactor designs, the amount of core distortion that will occur will not adversely affect core cooling.

QUESTION: Page 10

At what rate or rates are the rods assumed to heat up following LOCA and before reflooding begins?

ANSWER: Following the end of blowdown and until the bottom of the core is recovered, the fuel rods are assumed to heat up adiabatically. For the postulated double-ended break of a cold-leg pipe, the calculated clad heatup rate is approximately 40°F per second. This heatup rate considers both residual decay heat generation and the energy of metal-water reaction.

QUESTION: Page 10

To what extent is the steam buildup after blowdown taken into account in computing refilling and reflooding time? Explain how the steam generation is assumed to begin only after 20 secs, and the basis for your assumptions regarding the steam pressure and the direction of the steam once it begins to build up.

ANSWER: For the period of time following the end of blowdown and until the bottom of the core is recovered, no core heat transfer is assumed and therefore no pressure buildup from steam generation is calculated. Water is then assumed to refill about the bottom 20 inches of the core before there is any significant steam generation. During this initial refill, heat transfer from core is raising the reflood water temperature to its saturation temperature. This event was repeatedly observed during the FLECHT series of heat transfer experiments. At this time, approximately 30 seconds after the accident, appreciable steam generation begins. The quantity and direction of steam flow is based on a consideration of pressure gradients within the primary system and the containment, and the flow resistances of each of the parallel flow paths between the core and the containment. The driving force for steam flow from the core to the containment is the unbalanced hydraulic head between the downcomer and the core.

QUESTION: Page 10-11

Provide the details, including reference to supporting tests or analysis, upon which your conclusions on steam flow, water pressure, water and steam routes, are based.

ANSWER: The Staff has reviewed the Applicant's analysis of the refill and reflood portions of the accident and has concluded that the analysis was performed in accordance with Paragraph 7, Part 3, Appendix A of the Commission's Interim Policy Statement.

The Staff, both on a generic basis and for its safety evaluation of the Indian Point-2 plant reviewed the results from the FLECHT[1/] series of heat transfer experiments. The experimental work summarized in IN-1403 "Simulated Design Basis Accident Tests of CVTR Containment - Final Report," dated December, 1970, was considered by the Staff to determine containment pressure buildup following a LOCA.

1/ WCAP-7435, "PWR Full length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report" by J. O. Cermak et al dated January, 1970, and

WCAP-7544, "PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Group II Test Report" by F. F. Cadek et al dated September, 1970.

QUESTION: Page 11

Upon what assumptions, tests and analyses is the 80 second peak temperature based? Please identify all of those factors for which a 10% or less error would result in peak temperatures at any time in excess of 2300°F and the basis for the figures used for those factors.

ANSWER: As stated in Supplement 3 of the staff safety evaluation, the temperature calculations are based on AEC Interim Policy Statement of June 19, 1971. An exposition of the assumptions appears in that statement. The June 19 statement refers to a Westinghouse document, WCAP-7422, which in its current non-proprietary version further defines assumptions, tests, and analyses. The new calculations specifically referred to in Supplement 3 are presented in "Additional Testimony of Applicant Concerning ECCS Performance," July 13, 1971, Docket 50-247.

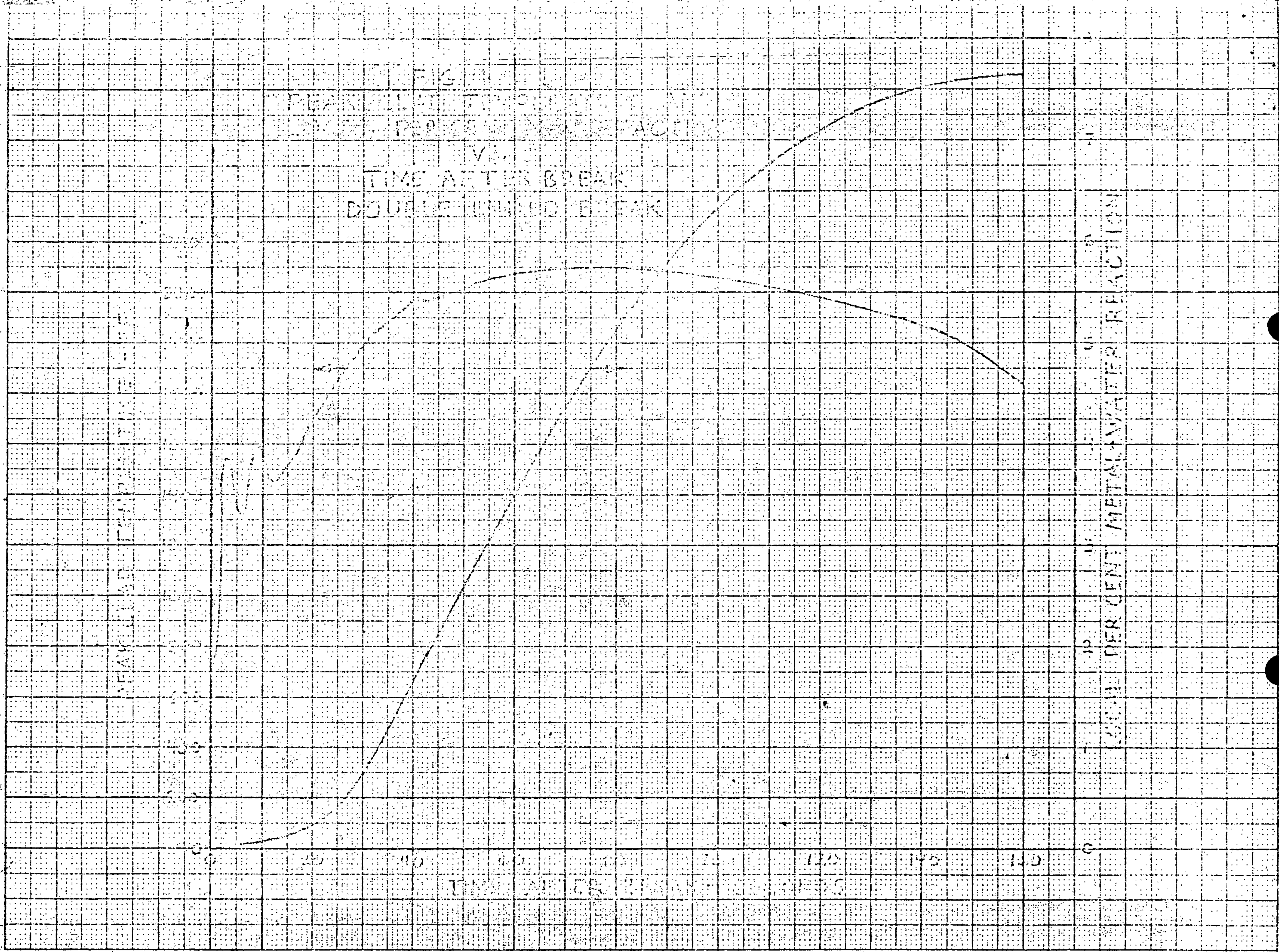
The 80 second peak temperature mentioned refers to the double-ended cold leg break calculation in the July 13 testimony. That temperature is at the 2300°F limit. Obviously, any error of even an incremental amount in the wrong direction would cause the calculated temperature to exceed 2300°F. It is important to point out that the parameters involved in these calculations already have error factors associated with them. For instance, a 20% factor is added to the proposed ANS decayed heat, a 2% calorimetric measurement factor is added to the power calculation, a 3% manufacturing tolerance factor is also added to the nuclear peaking factor. The calculated blowdown flow is reduced by 20% to account for possible error when applied to the fuel pin heatup calculation. Factors used in the codes which are "knowable" such as fuel pin power, physical properties, and dimensions are checked by the staff and have not been found to be in error. Comparative calculations for some cases have been made by the staff to check the thermo-hydraulic performance and fuel pin heatup. These checks have shown reasonable agreement with vendor calculations.

QUESTION: Page 12

Upon what tests and analyses is the total metal water reaction assumed to be less than 1%? In this regard please indicate the minimum temperature at which you assume metal water reactions will occur and the percent of reaction at that and higher temperatures up to 2300°F.

ANSWER: On page 19 of the July 13, 1971 testimony, the applicant states that less than 1% core metal-water reaction occurs for the limiting double-ended cold leg break. Westinghouse uses the LOCTA code which contains the metal-water model outlined in the July 13 testimony to calculated temperatures and metal-water reaction for 10 representative fuel elments in the core. A weighted average for these 10 elements is taken as the core-wide metal-water reaction.

The basic model used for metal-water reaction is the Baker-Just equation. This equation operates over the temperature range above 1800°F in LOCTA, but the calculated reaction is negligible below 1900°F. At that temperature with an initial oxide thickness of 0.2 mils the temperature rise rate due to metal-water reaction is only about 3°/second. The reaction rate goes up about a factor of 2 for each 100°F up to 2300°F for a given oxide thickness. The rate is inversely proportional to the oxide thickness. Since reaction rate is a function of oxide thickness as well as temperature, the temperature history as well as initial thickness are required to determine the amount of reaction which has occurred by a given time. A plot of percent reaction vs. time as calculated by the staff using the integrated Baker-Just equation is presented in Figure 1 based on the Westinghouse peak clad temperature transient for the double-ended cold-leg break.


VS.
TIME AFTER BREAK
PER CENT METAL-WATER REACTION

QUESTION: Page 12

What factors or combination of factors assumed in the computation of the metal water reaction would produce a metal water reaction in excess of 1% if there were an error of 10% in the calculated amount for the factor and upon what tests and/or analyses are those factors based?

ANSWER: It is virtually impossible to cause the core wide metal-water reaction to exceed 1% unless the hot spot reaches melting. If the hot spot transient is turned over at all before melting, the rest of the core transient is arrested much sooner. Therefore, the cooler portion of the core is contributing less and less metal-water reaction while the hot spot is still rising in temperature. Calculations of this phenomenon were made by the staff over a wide range of heat transfer conditions. The amount of core wide metal-water reaction never exceeded 1% even though the peak clad temperture exceeded 2300°F for several minutes. Westinghouse has independently arrived at the same conclusion.

QUESTION: Page 13

At what temperature do you consider fully irradiated (i.e. end of life) fuel rods to be immune from embrittlement failure following quenching and upon what tests and analyses do you base these figures? Please compare and comment upon the rod quench tests conducted by Westinghouse and discussed in the CCPE statement of issues (Para. 3.a.1.f).

ANSWER: On the basis of transient rupture of tests performed by ORNL and reported in ORNL-TM 3342 no differences resulting from irradiation other than a small and irregular reduction in ductility, were detected between unirradiated and irradiated fuel rod samples. On this basis the quench performance would be expected to be comparable for unirradiated and irradiated fuel rods. The criterion for shattering under quench conditions due to embrittlement would be associated with a time-temperature history rather than a single temperature criterion. For example, zero ductility at 1100°F would be associated with the following combinations of time-temperature; 2500°F for 2 minutes, 2400°F for 6 minutes, 2200°F for 15 minutes, 2000°F for 35 minutes, 1950°F for 35 minutes, according to Rittenhouse of ORNL. The limited Westinghouse data on irradiated Zr tubes reported in WCAP-7379 appear to fall within these limits and no shattering upon quenching from 2400°F, held for 1 1/2 minutes in air, was observed.

QUESTION: Page 15

Is a copy of the nonproprietary version of the Westinghouse June 1, 1971, ECCS report now available and if so, will you provide a copy?

ANSWER: Westinghouse requested that the document entitled "Additional Testimony of Applicant Concerning Emergency Core Cooling System Performance" submitted on July 13, 1971, as additional testimony to the Indian Point-2 plant, Docket No. 50-247, be considered as the nonproprietary version of the Westinghouse June 1, 1971 proprietary ECCS report.

QUESTION: <u>Page 17</u>

Explain the effect of permitting no steam flow in the intact loops during blowdown on the loss of accumulator water - i.e., at the intact loop inlet to the reactor what effect does the steam by-passing that inlet have on the rate at which accumulator water flows through the intact loop during blowdown and is swept out of the reactor? In particular, upon what tests and analyses is it determined that no more than 25% of accumulator water will be lost during blowdown?

ANSWER: Paragraph 5 of Part 3, Appendix A, of the Interim Policy Statement requires that all accumulator water injected during blowdown be lost and unavailable for refilling the reactor vessel and recovering the core. The amount of accumulator water injected during blowdown is determined from a consideration of the pressure difference between the accumulator tank and the point of injection in the primary system and the hydraulic resistance characteristics of the accumulator line. The assumption of no steam flow during accumulator injection is applied to conservatively limit steam venting flow paths during reflood and as such has no effect on accumulator injection rates. The amount of accumulator water discarded is primarily a function of blowdown time. Since blowdown time is a function of break size, the amount of accumulator water assumed to be lost will depend on break size. Table 4.2 of the Westinghouse report "Emergency Core Cooling Performance" dated June 1, 1971 presents additional data on the precent of accumulator water lost during blowdown as a function of break size. It should be noted that this amount of accumulator water discarded during blowdown does not include the complete spillage from the accumulator that is attached to the cold leg piping in which the break occurs.

QUESTION: B. With respect to Semiscale Tests 845 through 851 (June 29, 1971)

1. Page 1-1

Why cannot tests which check the adequacy of analytical models used in evaluating ECCS performance be directly applied to the performance of the reactor themselves?

ANSWER: The experimental information available from the Semiscale tests includes the time-variation of pressure, fluid temperature and density, metal temperature, and flow rate. By adjusting the ratio of break area to system volume, the Semiscale pressure behavior was reproduced on a real-time basis, and could be directly applied to the evaluation of the reactor performance. Other parameters could not be directly applied, however. For example, the steady-state core flow rate on a PWR is about 2×10^3 times the steady-state flow rate of Semiscale. The transient core flow rate on Semiscale could not be applied directly to PWR without some scaling factor, such as the 2×10^3 value. But the Semiscale design did not include as a basis the relative proportions of flow resistances around the flow loop. Nor did Semiscale have multiple loops, to serve as a core bypass. Hence the scaleup factor for Semiscale transient flow rate is not known.

It is in this context that the statement was made that the Semiscale parameters could not be applied directly.

QUESTION: 2. Page II-14

Compare the time in which rupture occurred in the tests to the time in which a double-ended pipe break would occur in a LOCA and explain the effect of the difference, if any, on ECCS performance.

ANSWER: The rupture time for the semiscale rupture discs was approximately 1-2 milliseconds, which is comparable to the postulated rupture time for a double-ended pipe break. This time is much shorter than the primary system blowdown and ECCS injection time and has an insignificant effect on the blowdown characteristics and ECCS performance.

QUESTION: 3. Page II-14

Compare the difference in accumulator water temperature in the tests to actual accumulator water temperature assumed for this plant and explain the effect of the difference, ia any, on ECCS performance.

ANSWER: The accumulator water temperature in the semiscale tests was approximately 140°F; the accumulator temperature at Indian Point 2 will be about 110°F. These temperatures are both significantly less than the primary system fluid temperature during the time of accumulator injection (490 - 290°F). Consequently, the small temperature difference between the two accumulator systems should have an insignificant effect on ECCS performance.

QUESTION #4: Page II-18

In the Indian Point No. 2 reactor is the water injected directly into the inlet plenum or does it enter the annulus? Compare this to the procedures used in the tests and explain how the differences affect the evaluation of ECCS performance.

ANSWER: On Indian Point 2 the accumulator water is injected into each of our cold-leg pipes which feed into the annulus. This mode was not effected on semiscale as there was only one cold-leg pipe, and it was broken. The expected reverse flow during the blowdown would negate the effect of an accumulator in a broken pipe, on a PWR. Consequently, in a PWR analysis, the accumulator water in the broken leg is assumed to be lost, as far as the core thermal consequences are concerned. There was no inlet annulus in semiscale, where no steady-state inlet flow rate would be directed downward. Instead, on semiscale, the inlet water was piped directly to the lower plenum. The procedures used in the semiscale tests, after test No. 845, were to deliver the ECC directly to the lower plenum.

We have calculated the thermal-hydraulic behavior of a three-loop PWR (not IP-2) with and without ECC delivery. The presence of the ECC makes little difference on the important parameters such as pressure, flow rate, and clad temperature prior to end-of-blowdown. On this basis we conclude that the differences in ECC injection modes have little effect on the evaluation of ECCS performance.

QUESTION #5: Page II-27

In what way, if any, do the pressure drop figures differ from the assumptions used in evaluating the ECCS performance for this plant and how do the differences affect the evaluation?

ANSWER: The system pressure drop for the single loop semi-scale was similar in proportion to the pressure distribution for a typical loop of a large PWR such as Indian Point II. By matching the pressure distribution, the experiment provided confirmation of the basic blowdown characteristics predicted by the blowdown codes (e.g., flow reversal through the core and flow stagnation in the steam generator for the postulated cold leg break). In the evaluation of Indian Point II, the appropriate pressure distribution and loop configuration were modeled for the computer code. Consequently, it is concluded that the pressure distribution characteristics predicted for Indian Point II plant is adequate.

QUESTION #6: P. III-4-5

What is the residual heat build-up for the fuel rods used in the semiscale tests? To what extent does the difference, if any, from the actual heat up rate of nuclear fuel rods affect the ECCS performance?

ANSWER: The stored energy capability of a reactor core is significantly greater than the semiscale heater rods. A typical reactor core such as Indian Point II contains the equivalent of five full power seconds of energy when operating at steady state full power. In addition, another full power second of energy would be generated during the postulated blowdown. Most of this energy would be transferred to the primary coolant during blowdown.

Because of the thermal properties of the semiscale heater rods, only a small amount of energy is stored in the heaters by comparison. To compensate for this difference and obtain the same net energy transfer to the coolant, the semiscale heaters were maintained at their steady-state full power for 6 to 8 seconds following the initiation of rupture. This mechanism would provide a reasonable simulation of the reactor core during blowdown, and would not result in a serious perturbation to the blowdown characteristics and ECC performance.

QUESTION #7: P. III-4-5

Explain in greater detail the cause of the ECCS failure during the semiscale tests.

ANSWER: An examination of the pressure-time curves for semiscale shows that the system pressure stayed above atmospheric pressure (to which semiscale was discharging) by an amount of 5-25 psi after ECC injection was complete. The pressure hangup was not seen on test 850, which had no ECC. A differential pressure of approximately 1.5 psi from the core outlet to the break is sufficient to eject ECC out the inlet pipe. Thus semiscale tests, with ECC, continued to blowdown after ECC injection was complete, in part due to the steam generated by heat transfer from hot metal surfaces. Due to the nature of the single loop design, the steam generated in the vessel would be vented preferentially up the inlet pipe, rather than up through the core and out the outlet pipe.

QUESTION #8: P. III-4-5

Compare the ratio of accumulator water to water in the reactor system for the semiscale tests to the comparable measurement in the reactor and explain why all accumulator water will not be lost during blowdown. In particular, discuss the projected rate of accumulator water flow for the semiscale tests with the actual rate of flow observed in the tests.

ANSWER: The injected mass of ECC on semiscale was sufficient, if none were lost, to recover the heater rods. On a PWR, in particular Indian Point 2, this is approximately true for the accumulator volume injected. All the water will not be lost on Indian Point 2 because:

(a) Indian Point 2 discharges to a containment instead of atmosphere. The containment back pressure would reduce the break flow to zero before a significant fraction of the ECCS has been delivered.

(b) A multiloop PWR design permits transfer of primary coolant from the intact steam generators and loops to the break location via the outlet plenum of the vessel, in addition to going through the core.

(c) The heat transferred from a PWR main coolant pipe varies directly as the pipe radius; the rise in fluid temperature varies inversely as the radius squared. Therefore the effect of heat transfer from hot (∿600°F) pipes during blowdown varies inversely with the radius (for the surface/volume ratio = $\frac{1}{r}$). The smaller pipe radius in Semiscale (4") as compared with a PWR (∿28") results in the heat transfer effect on Semiscale approximately seven times more significant on Semiscale. The end result is a higher fluid temperature in the pipe and loss of ECCS by vaporization. This same effect applies to the heat transfer from the vessel walls; a PWR vessel is on the order of 15 times larger in radius than Semiscale. Thus the Semiscale vessel heat transfer would be more significant.

(d) Semiscale could not scale elevation water heat in the annulus, which opposes ejection in ECC. A PWR annulus is about 16 feet high; semiscale had 3 feet in the inlet viser. Thus five times the pressure difference would be required on a PWR to lift water up.

The projected rate of ECC delivery as predicted by RELAP matched reasonably with the measured rate.

QUESTION #9: P. III-4-5

Explain what the computer codes being verified in these tests predicted would happen and what actually happen (sic) with respect to all observed phenomena where the results differed from the predictions.

ANSWER: The computer code being studied was RELAP 3. The following is a general summary of the comparison of code prediction vs. observation.

(i) The pressure prediction was reasonably good, until ECC came on, at which time the code predicted a more rapid decompression.

(ii) The core flow rate prediction was reasonably good.

(iii) The code predicted significantly higher heater rod surface temperatures than actually were measured.

(iv) The most significant difference between prediction and observation is in the density in the lower portion of the semiscale vessel. The code did not predict the low density, i.e., ECC ejection that occurred. The code does not have the capability for treating the heat transfer from the hot metal surfaces. (See answer to previous question).

QUESTION #10: P. III-7-14

Compare the results produced here with the predicted results and with the results now being predicted for this plant in the event of a LOCA.

ANSWER: The pages III-7-14 are data summary sheets for tests 845-851. Included are parameters such as initial system flow rates, temperatures, pressure, pressure drops, power, and ECC parameters. Final values are shown such as water remaining, time to DNB, decompression complete. Predicted results are not available for every test, and in any case, it is meaningless to compare many of the semiscale parameters to the Indian Point 2 parameters, due to the differences discussed in the answer to Question B-1.