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GENERAL ELECTRIC

NUCLEAR POWER

SYSTEMS DIVISION

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MFN-096-80

May 12, 1980

U. S. Nuclear Regulatory Commission
Division of Licensing
Office of Nuclear Reactor Regulation
Washington, D. C.

Attention: Darrell G. Eisenhut, Director
Division of Licensing

Gentlemen:

SUBJECT: ADDITIONAL INFORMATION CONCERNING NEDO-24708

Reference: Letter from D. F. Ross to T. D. Keenan, "Request for Additional Information Concerning NEDO-24708," November 6, 1979

The reference letter requested that the BWR Owners Group review and respond to several questions relative to the NRC review of NEDO-24708. The questions are primarily concerned with the response of the BWR to small break LOCA events complicated by loss of feedwater and severely degraded inventory control system response. Enclosed on behalf of the BWR Owners Group are 60 copies of the responses to the questions presented in the reference letter.

Please note that one figure (Page 23) of the enclosure is labeled proprietary to the General Electric Company pursuant to 10CFR2.790 and should be withheld from public disclosure. Also attached is an affidavit supporting the proprietary claim for this figure.

If you have any questions, please contact R. A. Hill (408) 925-5388.

RHBuchholz

R. H. Buchholz, Manager
BWR Systems Licensing
Safety and Licensing Operation

G006
s 1/60
CHANGE: L E
NRC PDR 1 NP
NSIC 1 NP

RHB:rf/657

Enclosure

cc: R. J. Mattson
L. S. Gifford

GENERAL ELECTRIC COMPANY

AFFIDAVIT

I, Glenn G. Sherwood, being duly sworn, depose and state as follows:

1. I am Manager of Safety and Licensing, General Electric Company, and have been delegated the function of reviewing the information described in paragraph 2 which is sought to be withheld and have been authorized to apply for its withholding.

2. The information sought to be withheld is contained in an attachment to a letter from Mr. R. H. Buchholz (GE) to Mr. Darrell G. Eisenhut (NRC), dated May 12, 1980, and consists of one page (page 23) entitled Critical Power at Low Mass Fluxes.

3. In designating material as proprietary, General Electric utilizes the definition of proprietary information and trade secrets set forth in the American Law Institute's Restatement Of Torts, Section 757. This definition provides:

   "A trade secret may consist of any formula, pattern, device or compilation of information which is used in one's business and which gives him an opportunity to obtain an advantage over competitors who do not know or use it.... A substantial element of secrecy must exist, so that, except by the use of improper means, there would be difficulty in acquiring information.... Some factors to be considered in determining whether given information is one's trade secret are: (1) the extent to which the information is known outside of his business; (2) the extent to which it is known by employees and others involved in his business; (3) the extent of measures taken by him to guard the secrecy of the information; (4) the value of the information to him and to his competitors; (5) the amount of effort or money expended by him in developing the information; (6) the ease or difficulty with which the information could be properly acquired or duplicated by others."

4. Some examples of categories of information which fit into the definition of proprietary information are:

   a. Information that discloses a process, method or apparatus where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

   b. Information consisting of supporting data and analyses, including test data, relative to a process, method or apparatus, the application of which provide a competitive economic advantage, e.g., by optimization or improved marketability;

c. Information which if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality or licensing of a similar product;

d. Information which reveals cost or price information, production capacities, budget levels or commercial strategies of General Electric, its customers or suppliers;

e. Information which reveals aspects of past, present or future General Electric customer-funded development plans and programs of potential commercial value to General Electric;

f. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection;

g. Information which General Electric must treat as proprietary according to agreements with other parties.

5. In addition to proprietary treatment given to material meeting the standards enumerated above, General Electric customarily maintains in confidence preliminary and draft material which has not been subject to complete proprietary, technical and editorial review. This practice is based on the fact that draft documents often do not appropriately reflect all aspects of a problem, may contain tentative conclusions and may contain errors that can be corrected during normal review and approval procedures. Also, until the final document is completed it may not be possible to make any definitive determination as to its proprietary nature. General Electric is not generally willing to release such a document to the general public in such a preliminary form. Such documents are, however, on occasion furnished to the NRC staff on a confidential basis because it is General Electric's belief that it is in the public interest for the staff to be promptly furnished with significant or potentially significant information. Furnishing the document on a confidential basis pending completion of General Electric's internal review permits early acquaintance of the staff with the information while protecting General Electric's potential proprietary position and permitting General Electric to insure the public documents are technically accurate and correct.

6. Initial approval of proprietary treatment of a document is made by the Subsection Manager of the originating component, the man most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within the Company is limited on a "need to know" basis and such documents at all times are clearly identified as proprietary.

7. The procedure for approval of external release of such a document is reviewed by the Section Manager, Project Manager, Principal Scientist or other equivalent authority, by the Section Manager of the cognizant Marketing function (or his delegate) and by the Legal

Operation for technical content, competitive effect and determination of the accuracy of the proprietary designation in accordance with the standards enumerated above. Disclosures outside General Electric are generally limited to regulatory bodies, customers and potential customers and their agents, suppliers and licensees only in accordance with appropriate regulatory provisions or proprietary agreements.

8. The document mentioned in paragraph 2 above has been evaluated in accordance with the above criteria and procedures and has been found to contain information which is proprietary and which is customarily held in confidence by General Electric.

9. Public disclosure of the information sought to be withheld is likely to cause substantial harm to the competitive position of the General Electric Company, and deprive or reduce the availability of profit-making opportunities. In addition, this information was developed at a substantial cost to the General Electric Company and its customers. The information is part of the General Electric technology base, and the precise value of the information is difficult to identify. However, this value is clearly substantial, and it would be a loss to GE if the information contained in Paragraph 2, above were disclosed to the public.

Glenn G. Sherwood, being duly sworn, deposes and says that he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 12th day of May, 1980.

Glenn G. Sherwood

Glenn G. Sherwood
General Electric Company

STATE OF CALIFORNIA)
COUNTY OF SANTA CLARA) ss:sem/985-37 40

Subscribed and sworn before me this 12th day of May 1980

Ruthe M. Kinnamon

NOTARY PUBLIC IN AND FOR SAID
COUNTY AND STATE

OFFICIAL SEAL
RUTHE M. KINNAMON
NOTARY PUBLIC · CALIFORNIA
SANTA CLARA COUNTY
My comm. expires MAR 29, 1981

175 Curtner Ave., San Jose, CA 95125

ANALYSIS GROUP

QUESTION 1. Provide a limiting analysis indicating the consequence of loss of all ECCS except two low pressure pumps. When would only two low pressure pumps be inadequate? For cases where two low pressure pumps are not adequate for core cooling, provide the time available to the operator for corrective actions (limiting case).

RESPONSE

The analyses submitted in Section 3.5.2.1 of NEDO 24708 (reference (1)) cover various break locations and sizes assuming one available low pressure coolant injection pump or low pressure core spray loop. The results of that analysis show that for any plant and any loss of inventory event, the availability of ADS and one low pressure coolant injection pump or core spray loop is sufficient to provide adequate core cooling if no high pressure injection is available. These analyses cover the case of multiple system unavailability due to postulated mechanical or electrical failures and/or operator error. The time available to the operator to depressurize the vessel and allow the low pressure system to inject was also shown, given the particular sequences of events analyzed, for cases requiring operator action.

QUESTION 2. Provide analyses where natural circulation is lost.

RESPONSE

All analyses presented in NEDO 24708 consider the effects of loss of natural circulation by including a natural circulation flow model in the SAFE code. This model simulates a loss of natural circulation flow if the calculated head difference between the levels of the liquid inside and outside the shroud is not sufficient to support a minimum flow. Further details of this modelling are given in NEDO 10329.

Because the effects of the loss of natural circulation are accounted for by the SAFE code, the analyses presented in reference (1) are adequate. The analyses show that, provided inventory is maintained in the vessel, natural circulation is inherent and is thus not lost. This is discussed further in Section 3.3 of NEDO 24708 and response to question III.2 of this submittal.

As long as the level in the downcomer is above the top of the jet pumps, positive natural circulation flow to the core via the normal recirculation path is assured. If the downcomer level falls below this point, the secondary natural circulation loop between the bypass and the core will still provide positive inlet flow to the core.

Depending on the size of a postulated break and the ECC systems available, the liquid flow at the core inlet can become zero or slightly negative after the liquid in the upper plenum has drained and the level in the bypass region has fallen to a low value. Effective core cooling after scram, however, does not rely on the direction of natural circulation flow but simply on the availability of inventory in the core. As long as water inventory exists in the core, the region below the mixture level will remain well cooled regardless of whether the flow at the inlet is positive or negative. The data to support this conclusion are reported in response to question III.2 of this submittal.

QUESTION 3. For all analyses conducted, in NEDO 24708, where operator action was required, provide the maximum time available for the operator to perform the required actions before exceeding the 2200°F limit (using 10 CFR 50, Appendix K assumptions). For any break size, including the no break case, what is the maximum time available to the operator for actuating the ADS? Consider with and without degraded conditions, including a stuck open relief valve.

RESPONSE

In the standard licensing analysis the only required operator action is to manually depressurize the vessel for cases when the high pressure systems fail and the ADS is not automatically initiated because a high drywell pressure signal is not present. Using Appendix K assumptions and assuming manual depressurization at 10 minutes results in peak clad temperatures which are substantially below 2200°F for all plants. Thus for all plants and for the analyses in NEDO 24708, the operator has at least 10 minutes to depressurize the vessel to prevent cladding temperatures in excess of 2200°F. Analyses using Appendix K assumptions and assuming manual depressurization after 20 minutes have been done for some plants in response to FSAR questions (LaSalle FSAR question 212.98 and Susquehanna FSAR question 211.90). These analyses show that the calculated peak cladding temperature does not exceed 2200°F for those plants and transients analyzed. The analyses submitted in Reference 1 show the core heatups to be slower using realistic assumptions. The time required for operator action to avoid reaching 2200°F clad temperature, using realistic assumptions, are therefore somewhat longer.

QUESTION 4. For BWR/4 and 5s, having Mark 2 containment, with automatic or manual LPCI diversion, provide the following analyses:

1) Limiting small break in the recirculation suction line in combination with all high pressure systems failed, and assuming LPCI diversion at 10 minutes.
2) Same as 1), without LPCI diversion.
3) Same as 1), plus the failure of the LPCS.
4) Same as 1), with early LPCI diversion, and
5) Same as 2), plus the failure of the LPCS.

For these analyses, provide the following transient curves:

a) Water levels (downcomer, lower plenum, core and upper plenum).
b) Vessel pressure.
c) Heat transfer coefficient (core and lower plenum wall heat).
d) Peak clad temperature.
e) ECCS, SRV, Core inlet, FW, ADS, and Steam line flow rates.
f) Lower plenum void fraction within the mixture level.

Provide a detailed description of the LPCI flow paths and discuss where the LPCI flow enters the lower plenum and at what elevation. Describe in detail, all the variables influencing the void fraction evaluation in the lower plenum.

Also describe the models used in determining the level (i.e. wall heat transfer models, bubble rise model, treatment of determining voids due to depressurization).

RESPONSE

An analysis assuming LPCI diversion at 10 minutes was presented in NEDO 24708, Section 3.5.2.1 (reference (1)). It shows the effect of diverting up to two RHR pumps from the LPCI mode to the containment spray mode at approximately 10 minutes after the accident. This analysis shows that only very small breaks are affected by diversion (i.e., breaks for which the reactor core does not reflood before 10 minutes). The limiting break/failure combination is a break in a core spray line with only one LPCI remaining after diversion.

Section 3.5.2.1.2 shows the system responses for the case of LPCI diversion following a recirculation line break. When compared to the recirculation suction line breaks, it is seen that the system responses for LPCI diversion are very similar. The cases in that section with one LPCI available are typical of the system responses expected given LPCI diversion to containment spray at 10 minutes.

Section 2 of NEDO 24708 provides a detailed description of the LPCI flow paths. For BWR/3 and BWR/4 plants the LPCI flow enters the lower plenum through the jet pumps. For BWR/5 plants the LPCI flow enters the lower plenum through the bypass leakage flow paths.

An analytical description of the SAFE code is contained in NEDO 10329. Paragraphs B.3.10, B.3.11, and B.3.15 describe the level calculations, bubble rise model, and wall heat transfer models.

The effect of ECC water injection below the mixture level is provided in the response to question I.2.

RESPONSE TO #4 (Cont'd)

The parameters to be plotted with each analysis were agreed to in a July 12, 1979, meeting between the BWR Owners' Group, GE, and the Staff. Several of the parameters which were requested in Question 4 were not provided because they are not available as output variables from SAFE. These include the core inlet flow rate and separate water levels for the lower plenum, core, and upper plenum. The SAFE code does not calculate separate water levels for these three regions, but treats them as one region to calculate a level inside the shroud.

The Appendix K core heat transfer assumptions were used in this analysis with two exceptions which are identified in Section 3.1.1.3.3.1 of NEDO 24708. The exceptions are maintaining nucleate boiling to a local void fraction of 0.99 and increasing the core spray heat transfer coefficient from 4 to 12 BTU/hr ft^2 ^{o}F. The lower plenum wall heat transfer coefficients were not printed out, but section B.3.10 of NEDO-10329 describes the calculation of the vessel wall heat transfer in SAFE.

QUESTION 5. Discuss all information available to the operator which will detect core uncovery.

RESPONSE

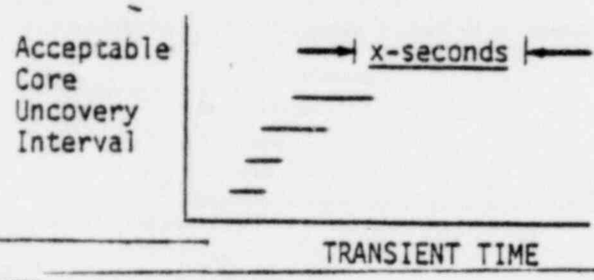
Reference 2 provides a discussion of the information available to the operator which will detect the approach to inadequate core cooling and core uncovery.

QUESTION 6. What is the system response of BWR classes, other than the BWR/4-218, to the "no break" case? How much time is available for manual ADS?

RESPONSE

The system response to a no break case for all product lines is extensively discussed in References 1 and 3. Time available for operator action was discussed in response to Question 1 of this submittal.

QUESTION 7. What length of time is core uncovery acceptable before the 2200 degree limit (using 10 CFR 50, Appendix K assumptions) is exceeded? Provide a chart of acceptable core uncovery time as a function of when the uncovery begins, i.e.:



RESPONSE

The analysis provided in Section 3.5.2.1 of NEDO-24708 shows the core uncovery time and associated cladding heatup under several conditions using models more realistic than Appendix K models. The core uncovery time for those cases depends on the break type, depressurization rate, and steam generation rate in the core in addition to the time of core uncovery.

As suggested by the chart in the question, the initial time of core uncovery is the main factor which determines rate of cladding heatup, and thus the amount of core uncovery time to reach a given cladding temperature; however, there are several other factors which govern the cladding temperature when Appendix K assumptions are used. These include features unique to the plant being considered, the ECC systems which are assumed to be available during the postulated accident, and the power history of the plant prior to the postulated accident. It is not possible to draw a single chart which applies to all plants and to all conditions and combinations of available systems.

For a given plant, an estimate of the time available to reach a given cladding temperature as a function of time of core uncovery using Appendix K assumptions can be made using the FSAR analysis. The FSAR analysis contains water level and PCT plots for several break sizes and types. These analyses can be used to generate a chart similar to that requested. However, since the PCT for most breaks is below 2200°F, only the time available to reach some lower cladding temperature can be determined.

QUESTION 8. Conclusion 6 in section 3.1.1.1.2.2.4, states that the core level will always remain higher than the downcomer level for small break LOCAs. This statement appears true if CCFL is not considered. Comparisons of Figures 3.1.1.3-5 with 3.1.1.3-6 show that CCFL can create a higher downcomer level versus the core level. With the arguments presented, the staff can not agree with the conclusion that CCFL plays a negligible role in a small break LOCA. Provide additional justification for the neglect of CCFL and discuss the implication of hot channel behavior, where CCFL may play a crucial role. For every analysis where the core uncovered, provide a plot of coolant velocity exiting the core as a function of time and discuss at what velocities CCFL will begin to play a role in preventing or retarding liquid penetration.

For all cases where CCFL can retard the penetration of ECC, expand your analysis by conducting a REFLOOD evaluation, which models the CCFL phenomenon.

RESPONSE

The SAFE code was used in the analyses to demonstrate the relative differences in the overall reactor response for various break and failure combinations, as discussed in the Owners' Group/GE meeting with the NRC staff on July 12, 1979. The analyses were not intended to be used for calculating the exact consequences of every situation analyzed, but for preparation of operator guidelines where management of total vessel inventory is the objective. The most important variable for determining the relative differences between events is the total vessel inventory, and to a lesser extent the exact distribution of the inventory in the vessel. The exact inventory distribution between various regions impacts the calculated consequences (i.e., the calculated peak cladding temperature) and is to some extent affected by CCFL, as shown in the Figures referred to in the question. The Figures and the accompanying explanation, however, also show the effect of CCFL to be transitory and not of importance to the management of total vessel inventory, as intended in the guidelines. The effect of CCFL on water level measurement is discussed in Reference 2.

QUESTION I.1. Provide assurances that the 10 CFR 50.46 criteria (using 10 CFR 50, Appendix K assumptions) will not be exceeded for all analyses in NEDO 24708, which resulted in core uncovery.

RESPONSE

The great majority of the analyses in NEDO 24708 did not result in core uncovery. The cases for which uncovery occurred for a brief period of time are shown in Table I.1-1. Using realistic assumptions none of these cases approach 10 CFR 50.46 criteria.

TABLE I.1-1

PLANT	FIGURE GROUP	BREAK	SYSTEMS OPERABLE
BWR/2	3.1.1.1-19	0.1 ft^2 suction	All
BWR/2	3.1.1.1-50	0.1 ft^2 suction	Low pressure
BWR/4	3.1.1.1-27	0.1 ft^2 suction	Low pressure + RCIC + CRD
BWR/4	3.1.1.1-38	0.2 ft^2 suction	Low pressure
BWR/4	3.1.1.1-46	0.1 ft^2 suction	Low pressure + feedwater
BWR/4	3.1.1.1-49	0.1 ft^2 suction	Low pressure
BWR/4	3.1.1.1-51	0.1 ft^2 steamline	Low pressure
BWR/4	3.1.1.1-52	0.5 ft^2 outside steamline	Low pressure
BWR/6	3.1.1.1-40	0.1 ft^2 suction	HPCS + low pressure

With 10 CFR 50 Appendix K assumptions, the most severe small break is typically a recirculation suction break of about 0.07 ft^2 with a single failure of a high pressure ECC system (EC, HPCI, or HPCS), leaving only the ADS and low pressure systems available. The maximum peak clad temperatures (PCT) calculated for the 0.07 ft^2 suction break using 10 CFR 50, Appendix K assumptions are listed in Table I.1-2 and show that the 10 CFR 50.46 criteria are not exceeded.

Each of the cases from NEDO 24708 identified in Table I.1-1 has at least the ADS and all of the low pressure ECC systems available and in some cases a limited number of high pressure systems also available. Consequently, the cases identified above will be no more limiting than the licensing submittal results. Smaller or larger breaks than .07 ft^2 yield lower PCTs than those shown in Table I.1-2 and demonstrates that the analyses of NEDO 24708 also do not exceed the 10 CFR 50.46 criteria.

TABLE I.1-2

PLANT	BREAK	PCT (^oF)
BWR/2	0.07 ft^2 suction	2200 (limiting case)
BWR/4	0.07 ft^2 suction	1260
BWR/6	0.07 ft^2 suction	1670

QUESTION I.2. Describe how the SAFE computer program evaluates the internal system pressure gradients which result from liquid flashing within the core and steam condensation due to ECC injection. Are internal system flow rates affected by these pressure gradients? If the code does not model such phenomenon, provide justification for its neglect, including experimental verification.

RESPONSE

The SAFE code is used for long term inventory calculations. In SAFE, internal pressure gradients are based on the density head in the downcomer and inside the shroud. The short term blowdown model, LAMB, calculates a more detailed pressure distribution accounting for all components of pressure drop during rapid depressurization transients when the pressure distribution within the vessel is significant. During the long term transient, the density head in the downcomer is the dominant component of pressure drop.

When subcooled ECC water is injected into the vessel, internal pressure gradients are slightly different depending on whether the injection is below or above the existing mixture level. If subcooled water is introduced below the mixture level, vapor below the level condenses and the region becomes subcooled with a small layer of saturated liquid in contact with vapor at the free surface. In this situation the degree of local depressurization is very slight because of the presence of a liquid continuum.

If subcooled water is injected into steam, rapid condensation of steam occurs as steam is drawn to the injection point from within the region. The extent of local depressurization depends on the resistance between the injection region and adjacent regions. For core spray injection, the low resistance path for steam through the steam separators precludes large internal pressure differences. The calculation of one thermodynamic pressure for the vessel is adequate, because these local variations are small (less than one inch of head) and have little effect on the overall results.

The significant effects of subcooled injection are bulk subcooling of the upper plenum and CCFL collapse, which are unaccounted for in the SAFE code. Both of these effects greatly accelerate lower plenum refilling and core reflooding. Similarly for LPCI injection into one bank of jet pumps, the other jet pumps provide an open path between the lower plenum and the downcomer. The opening at the top of the core bypass region is also sufficiently large to prevent large pressure differences between the bypass and upper plenum. Local depressurization in the bypass region will hasten the filling of the bypass. Thus the omission of these effects is conservative from the standpoint of calculating vessel inventory.

Experimental data from the Two Loop Test Apparatus (TLTA) support these conclusions. Local pressure differences between different regions following ECC injection show negligible variations between the steam regions. No dramatic changes in flow rates or direction are observed.

QUESTION I.3. Provide assurances that the break size and the assumptions chosen in the analyses result in the most severe test for operator actions.

RESPONSE

Pipe breaks inside containment require no operator action. Pipe breaks outside containment with high pressure injection systems unavailable require manual depressurization of the reactor to allow low pressure systems to inject. This action is not dependent on break size.

The effects of the assumptions used in the analyses on the results is answered by the discussion in Section 3.1.1.3 of NEDO 24708.

<u>QUESTION II.1.</u> Section 3.3.2.2 states that application of recirculation pump flow during a LOCA would require additional evaluations before a recommendation can be made to restart the pumps. Verify that GE has the analytical capability to model such transients. Provide verification to show such analyses are valid.

<u>RESPONSE</u>

The calculation of Section 3.3.2.2 was intended to quantify approximately the effects of restarting the recirculation pumps. Section 3.3.2.5 of NEDO-24708 identifies a number of reasons for not recommending recirculation pump restart. These include concern that pump restart will detract operator attention from operation of inventory makeup systems to maintain water level, will decrease reliability of emergency power sources due to increased electrical demand from the recirculation pump motor, and will be less beneficial to plant performance for BWR/3 and BWR/4 plants than injection of LPCI flow through the recirculation system. Further analyses of recirculation pump operation are presented in Reference 1. These analyses show that continued operation of the recirculation pumps results in little change in the time available for operator action. Considering all factors, restart of the recirculation pumps is <u>not</u> recommended. It was not the intention of Section 3.3.2.2 to suggest that a recommendation for recirculation pump restart was contemplated.

QUESTION II.2. For non-jet pump plants, provide test results which verify that only one recirculation loop provides sufficient communication between downcomer and core to provide natural core circulation.

RESPONSE

There are no direct test results to show that only one recirculation loop (on an external loop non-jet pump plant) can provide sufficient communication between downcomer and core to provide natural core circulation. However, an incident at the Oyster Creek plant provided strong assurance that this is the case. The results of an analysis of the Oyster Creek incident were submitted to the staff in Reference 4. This submittal shows that with one loop open, conditions sufficient to support normal circulation would have existed.

QUESTION III.1. What conditions will result in negative core flow? How does this influence natural circulation?

RESPONSE

Refer to response to Questions 2 and III.2 of this submittal.

QUESTION III.2. Section 3.3.1.3 addresses the need for natural circulation in a BWR. In that section, reference was made to an experiment which showed that no dryout will occur, even at zero inlet flow, provided water is available in the upper plenum. Document the experiment, provide a graphic description of the experiment, discuss what break size will result in loss of upper plenum inventory within 10 seconds of scram, and the applicability of the experimental geometry to all operating BWR plants.

RESPONSE

The experiments referenced were full-scale, full-power critical power tests of prototypical bundles in the ATLAS loop. A large data base exists for positive mass fluxes down to 0.025×10^6 lb/hr-ft^2. The data from different fuel assemblies show that below a mass flux of 0.1×10^6 lb/hr-ft^2, the effects of local peaking and bundle geometry details become negligible. Refer to Figure III.2-1. Assemblies 32B and 33A are standard 8x8 assemblies with one water rod and 63 fuel rods of 0.493" diameter. Assembly 50C is of the BWR/6 design while Assembly 52A represents the BWR/2 through 5 retrofit geometry, both with two 0.591" diameter water rods and 62 0.483" diameter fuel rods. The four assemblies shown represent a variety of local peaking patterns and spacer types. Results plotted in Figure III.2-1 show that at mass fluxes of 0.1×10^6 and lower the data points are indistinguishable. This establishes that the flow results are valid for all 8x8 fuel assemblies in the field, and there is no reason to suppose that the results for 7x7 assemblies would be otherwise.

Critical power tests have been run at zero and small negative inlet mass fluxes. Figure III.2-2 shows schematically the test bundle and instrumentation for this test. In this test the BWR/6 bundle exit geometry was simulated very closely and an actual BWR/6 upper tie plate was used. The heater rod has a uniform axial heat flux shape with a maximum rod local peaking of 1.232. The top of the test section was connected to a direct contact condenser, thus ensuring that water was available in the test upper plenum. Liquid downflow was achieved by venting liquid from the bottom of the test bundle to atmosphere.

In operation, test section flow was established and bundle power was increased in steps of 5 to 20 KW (allowing 2 to 10 minutes for system stabilization between steps) until a boiling transition was observed as indicated by heater rod thermocouples.

Results from this test are also shown in Figure III.2-1 and are seen to fair into the positive flow curves in a continuous manner.

Figure III.2-3 shows a more detailed plot of the downflow and zero flow data. A range of pressure from 1000 - 300 psia showed a relatively small effect on the critical power. At zero net flow, the critical power is of the order of 650 - 680 KW. For net downflow (negative core flow), the critical power drops and levels out at about 300 KW for a downflow mass flux of 0.125 x 10^6 lb/hr-ft^2 at 1000 psia. At larger downflows the critical power would be expected to increase.

Approximately 10 seconds following scram, the decay power level in a BWR drops below 300 to 400 KW per bundle and thus no boiling transition is expected following this time as long as the bundle remains covered. Even for a DBA, upper plenum inventory will not be depleted within 10 seconds of scram: for smaller postulated breaks, the bundle remains covered longer.

It should be recognized, of course, that in the first few seconds after the scram the decay power is higher than 400 KW and a boiling transition can occur depending on the flow coastdown in this period. The licensing basis (Appendix K) calculations account for this possibility. Figure III.2-4 schematically illustrates this situation. Figure III.2-4(a) shows typical heat flux history corresponding to the decay power level following scram. This is essentially independent of the break size. Figure III.2-4(b) shows the critical power vs. flow rate as obtained from previously described data (i.e. Figure III.2-1 and -3). From these two curves, a "critical flow rate" vs. time can be cross plotted, above which no boiling transition will occur (Figure III.2-4(c)).

The flow coastdown history depends on the break size. Figure III.2-4(c) shows the nature of the flow coastdown as a function of increasing break size. The comparison of actual flow rate vs. "critical flow rate" is made on a local basis utilizing transient computer codes.

For small break sizes the local flow rate always lies above the "critical flow" rate, so that no boiling transition occurs. For intermediate size breaks, a boiling transition occurs in the upper core region which is a low power region. This region is subsequently rewetted during lower plenum flashing. For very large breaks, the dryout region may propagate further into the bundle prior to lower plenum flashing. Tests in the Two-Loop Test Apparatus conclusively show rewetting of the entire bundle and no sustained heatup is observed until total core uncovery, some 30-40 seconds into the DBA transient.

The analysis of large and intermediate size breaks is performed using the short term transient codes LAMB and SCAT which account for the LOCA flow history and employ the GEXL correlation for the calculation of boiling transition. The assumption of nucleate boiling until uncovery is only made for small breaks.

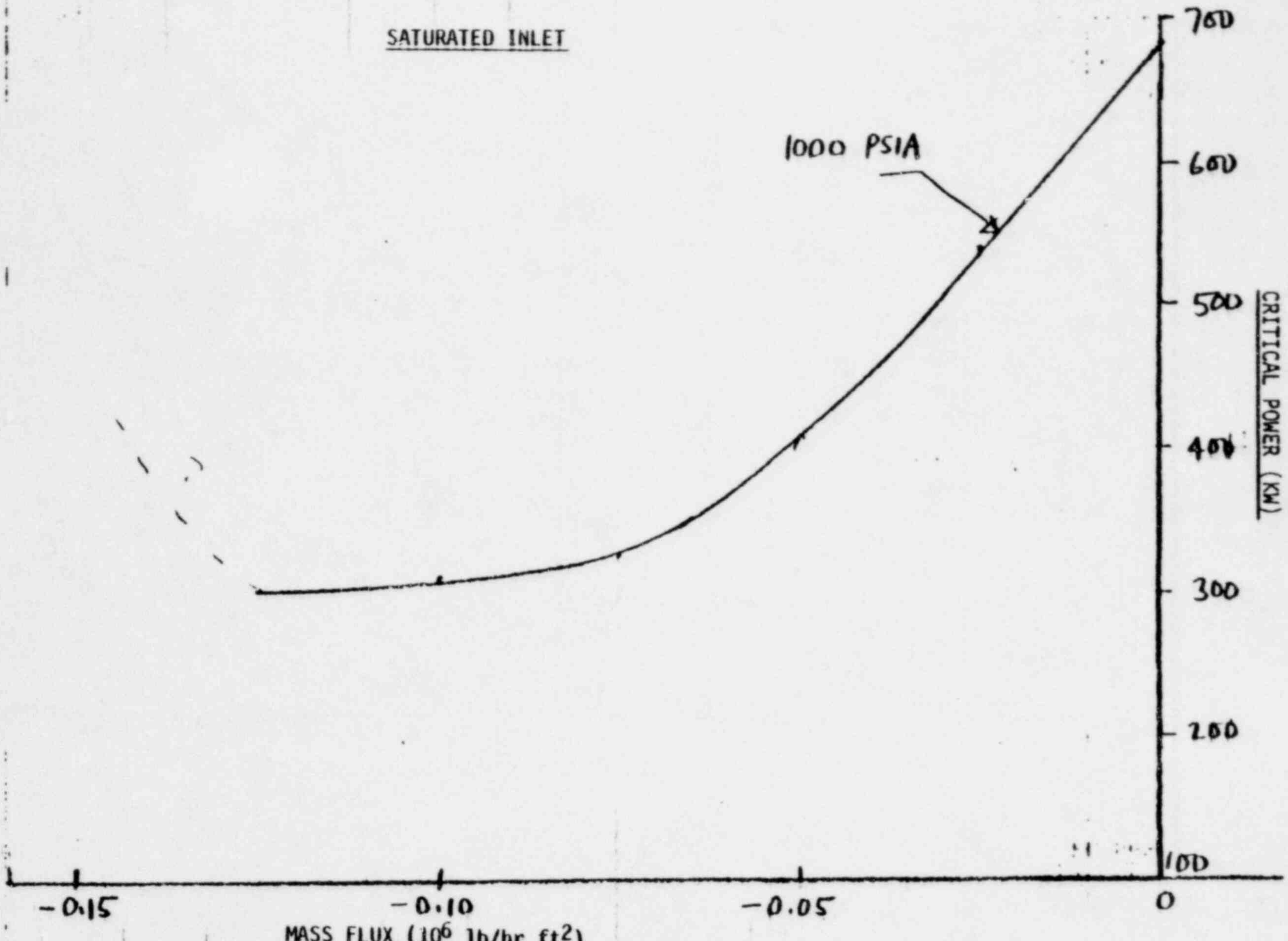


FIGURE III.2-3; CRITICAL POWER AT NEGATIVE MASS FLUXES

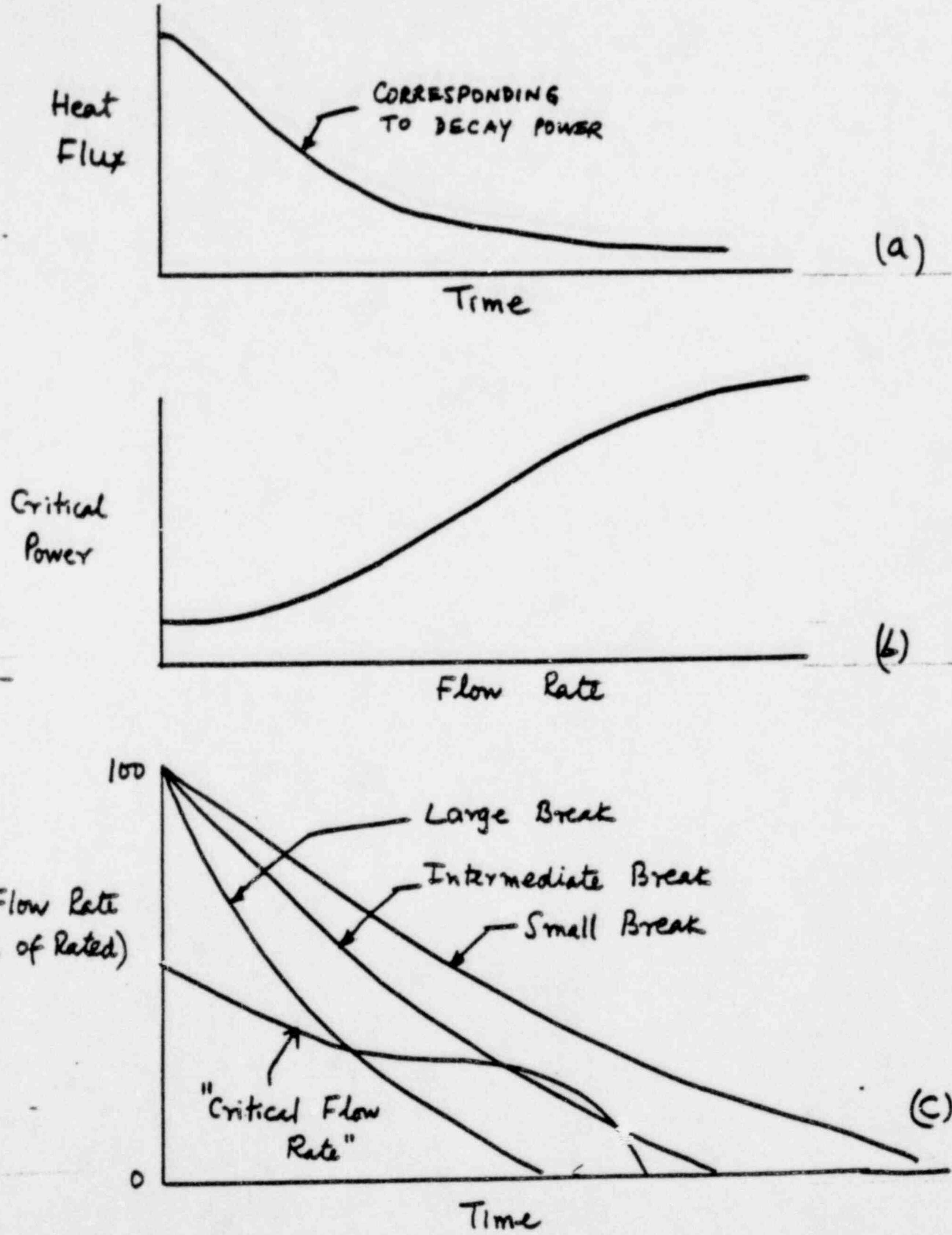


FIGURE III.2-4 OCCURRENCE OF BOILING TRANSITION DURING FLOW COASTDOWN

QUESTION IV.1. Analyses conducted utilizing the SAFE computer code showed numerical instabilities or non-convergence in evaluating model level and flows: i.e.

Figures 3.1.1.1-19.2
-19.5
-20.2
-20.5
-47.5,etc.

Discuss, in detail, the reasons for these oscillations and verify their impact on the analytical results. Are inertial effects accounted for in the analytical equations? If not, how does the neglect of inertia influence the system response? Verify that the code has converged to a solution for the analyses performed.

RESPONSE

The "oscillations" observed in Figures 3.1.1.1-19.2, -29.5, -20.2, -20.5, and -47.5 are not due to numerical instabilities or non-convergence but to certain modelling assumptions which apply past the time period of concern for the analysis. The "oscillations" occur after the time the core is adequately cooled and do not impact either the analytical conclusions of NEDO 24708 or licensing calculations.

The water-level oscillations in Figures 3.1.1.1-29-2, -20-2, etc. occur because when vessel level has restored to the "spillover" region of the steam separators, the SAFE model assumes saturated rather than subcooled conditions. To demonstrate this, the discrepancy in spillover specific volume was corrected so that a consistent specific volume for the upper vessel liquid was used. The results of the new calculation are shown in Figure IV-1.1. In this figure the level inside the shroud is seen to rise to the spillover elevation of the steam separator and no oscillations occur.

The recirculation flow rate oscillations in Figures 3.1.1.1-19-5, and -20.5, etc. directly result from the water level oscillation just discussed. The natural circulation flow rate is related to the hydrostatic head inside the shroud, which in turn depends on the in shroud water level. The results of the calculation with the discrepancy in specific volume corrected are shown in Figure IV-1.2, and as can be seen the oscillations are no longer present.

The flow rate oscillation in Figure 3.1.1-47.5 does not represent a numerical instability or non-convergence but results from the lack of modelling inertia effects in SAFE. Neglecting the inertia terms (especially for small breaks) has no significant impact on the overall inventory calculation, which is based on average flow characteristics.

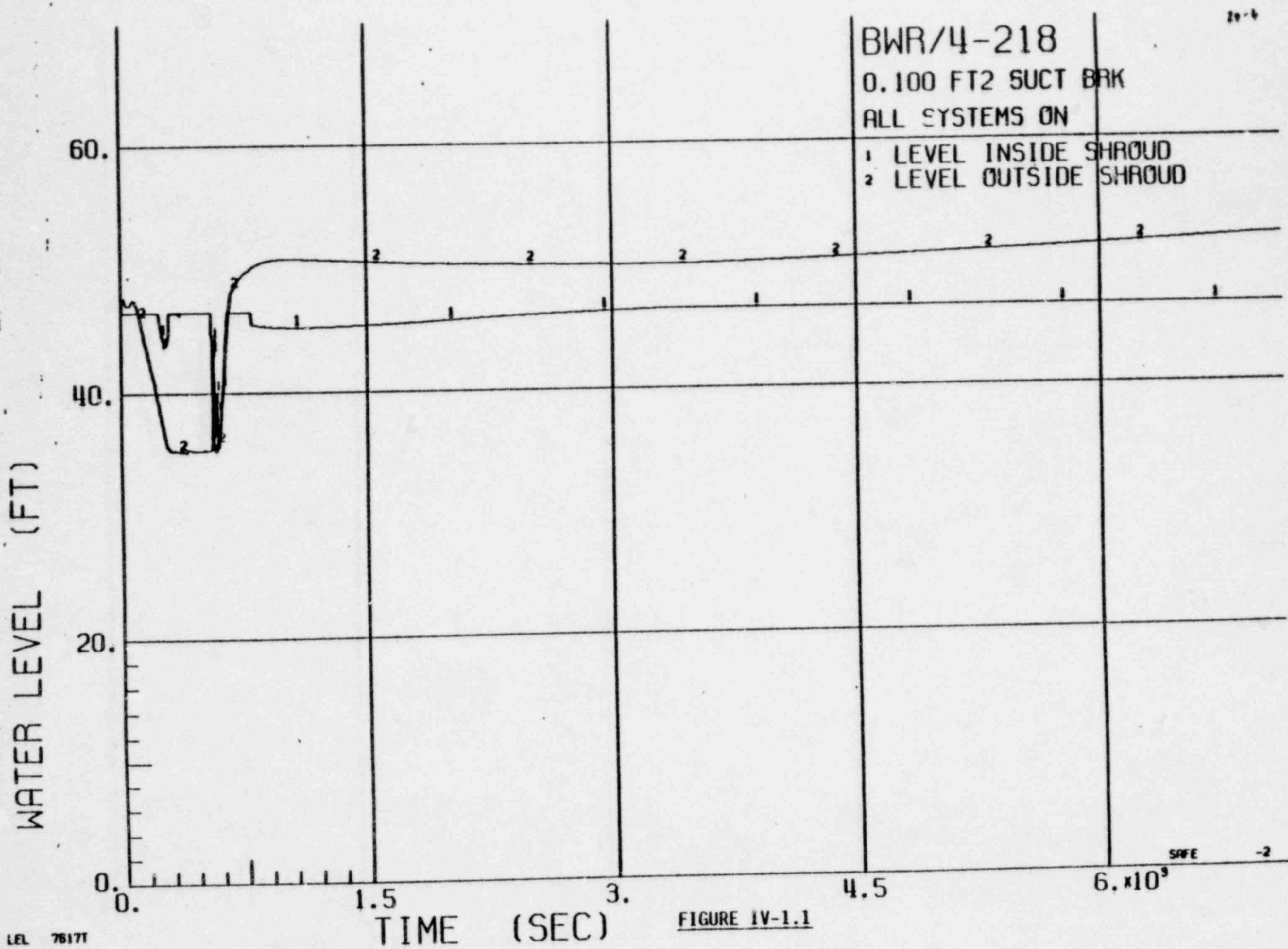


FIGURE IV-1.1

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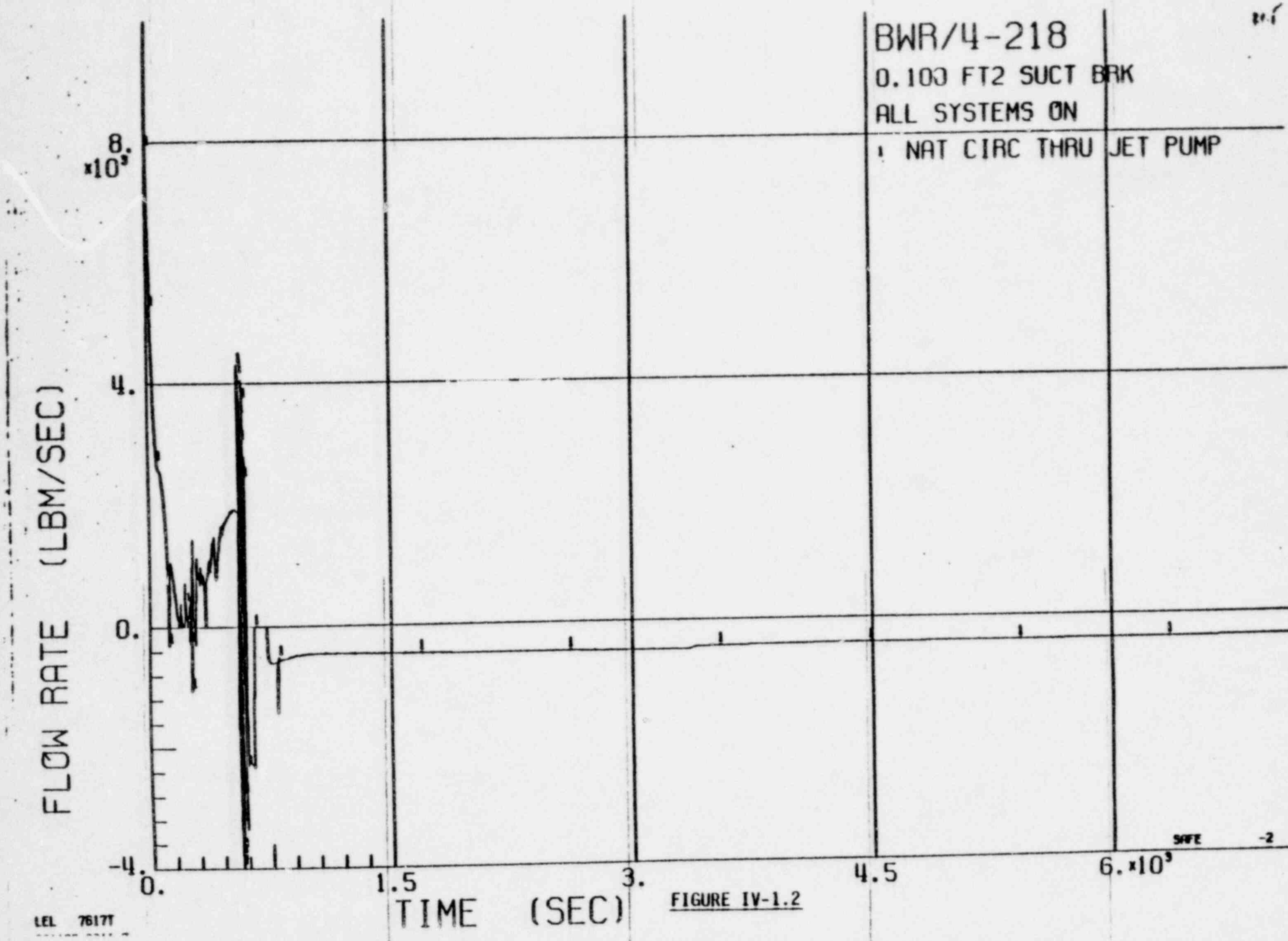
BWR/4-218
0.100 FT2 SUCT BRK
ALL SYSTEMS ON
1 NAT CIRC THRU JET PUMP
FLOW RATE (LBM/SEC)
8.
x10³
4.
0.
-4.
0.
1.5
3.
4.5
6. x10³
TIME (SEC)
SAFE
-2
LEL 7617T

FIGURE IV-1.2

QUESTION IV.2 Figure 3.1.1.1.-19.4 and -38.4 indicate the ADS system opening and closing. Is the on-off behavior of the ADS system an automated process? How many cycles can the ADS accommodate? Provide the settings for the on-off logic. If the ADS valves do not cycle, describe in detail why the cycling is observed in the mentioned figures.

RESPONSE

The cycling of the ADS valves observed in Figure 3.1.1.1.-19.4 (BWR/2) and 3.1.1.1.-38.4 (BWR/4) is due to the cycling of the main disc springs, which close the valve at a pressure of about 50 psig and allow it to open again at about 100 psig. The solenoid of the ADS valve remains open during the cycling, so there is no limit to the number of cycles due to air supply.

The BWR/6 ADS valve is of a direct-acting design, so the valves do not cycle after opening.

QUESTION IV.3 Discuss the worst case power profile for small breaks. What power profile was used in the analyses?

RESPONSE

The power profiles used in the analysis are the design axial power shapes as described in NEDO-20566. These are also used for the Appendix K analysis. Paragraphs I.2 and C.4.2 of NEDO-20566 describe the results of a sensitivity study performed using the LOCA analysis computer codes which concluded that the design axial power shape is appropriate for use in evaluating the consequences of a LOCA and that the total energy transferred to the coolant is much more important in determining the overall system response than the power shape.

QUESTION IV. 4 Analyses indicate that part of the operator's responsibility is the requirement to throttle the ECC systems. List all conditions for such operator action, and describe, in detail, the reasons for such actions.

RESPONSE

The conditions for which manual control of ECCS is recommended will be set out in the Emergency Procedure Guidelines. For example, it is recommended that HPCI and RCIC be throttled to maintain the reactor water level below the high-level trip points for these systems. This action stabilizes water level, avoids cycling of the systems, and avoids the possibility of their being put out of service due to water carryover in their steam supplies.

QUESTION IV.5 What information does the operator have, which assures him of adequate NPSH for his ECC system?

RESPONSE

The ECC systems (ECCS) suction lines, pumps, and strainers are designed and arranged (assuming the suction strainer 50% plugged) such that adequate net positive suction head (NPSH) is available to the pumps for the most limiting short term post-LOCA conditions.

Pre-operational tests are performed to verify that the NPSH design requirements are met. Routine surveillance tests and maintenance are performed during normal plant operation as required to assure that the design NPSH is maintained. Therefore, adequate NPSH is assured by proper design, test and maintenance.

There is no direct indication of available NPSH in the control room. Suppression pool temperature and system flow conditions, however, are available in the control room. These can be compared with design values to ensure adequate NPSH.

REFERENCES

(1) Letter, Buchholz to Ross, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors", (Prepublication form of NEDO 24708, Section 3.5.2.1), November 30, 1979.

(2) Letter, Buchholz to Ross, "Bulletins and Orders Task Force Request for Information", (Prepublication form of NEDO 24708 Section 3.5.2.3), December 28, 1979.

(3) Letter, Buchholz and Keenan to Ross, "NUREG 0578 Requirement 2.1.9 - Implementation by Owners' Group", MFN-074080, March 31, 1980.

(4) Letter, IR Finfrock, Jr. to Director, Nuclear Regulatory Commission, "Oyster Creek Nuclear Generating Station, Docket 50-219, Transient of May 2, 1979", May 12, 1979.