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MINUTES OF THE
ACRS SUBCOMMITTEE MEETING ON
EMERGENCY CORE COOLING SYSTEMS
MARCH 25, 1980
WASHINGTON, DC

The ACRS Subcommittee on Emergency Core Cooling Systems (ECCS) held a meeting on March 25, 1980, at 1717 H Street, N.W., Washington, D.C. The purpose of this meeting was to review the adequacy of Westinghouse Upper Head Injection (UHI) ECCS model for small break Loss of Coolant Accidents (LOCAs) and also to discuss several ACRS generic items related to Reactor Coolant Pump Overspeed and the ECCS Capability for older, current and future plants. Notice of this meeting was published on March 11, 1980, in the Federal Register, Volume 45, Number 49; a copy is included as Attachment A. Dr. Andrew Bates was the Designated Federal Employee for the meeting. A list of meeting attendees is included as Attachment B.

EXECUTIVE SESSION

Dr. Plesset, the Subcommittee Chairman, convened the meeting at 8:30 a.m, reviewed briefly the schedule for the meeting and solicited preliminary remarks from the Subcommittee and its consultants on the subject matter. Dr. Catton raised the following questions to be answered during the course of the meeting:

1. In view of the fact that dumping a large amount of UHI water on top of a partially voided core may lead to 3-Dimensional fluid motion, is there any evidence that justifies the validity of the 1-Dimensional drift flux model which is used in the UHI small break LOCA calculations?
2. What is the model used by Westinghouse in calculating the froth level in the core?; what is the justification in selecting that specific model?

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3. In light of the fact that UHI actuation may not only halt the single phase natural circulation but may even reverse its direction, what is the peak temperature reached during the period at which normal natural circulation reestablishes itself?

Dr. Schrock expressed concern about the validity of the Westinghouse small break UHI LOCA calculations that are based on 1-Dimensional drift flux model. He also commented that the idea of defining the operator actions to handle transients based on certain fictious calculations does not seem to be a valid approach; he believes that operator action can be best determined through best estimate assessment of the actual situation.

Dr. Theofanous questioned the validity of the use of some analytical calculations to define the expected system behavior and hence the operator action. He commented that he is not confident that such calculations will adequately represent the actual system behavior under transient conditions.

Dr. Zudans commented that he has difficulty in accepting one of the conclusions listed in WCAP-9639 (Report on Small Break Accidents for Westinghouse Steam Supply System with Upper Head Injection) which states that water levitation above a partially uncovered core is not indicated by either drift flux approach. He believes that this conclusion is based on assuming uniform velocity over the entire cross section of the core. He asked whether there is any assurance that such an uniform velocity will exist throughout the entire cross section of the core and hence steam and vapor velocities are not high enough to levitate the water.

PRESENTATION BY WESTINGHOUSE - DR. DOCHERTY

Dr. Docherty from Westinghouse reviewed briefly the main functions of the UHI system indicating that during small break transients the UHI system:

1. delivers a nominal water volume of 1000 ft^3 at an actuation pressure of 1250 psia,

2. delivers water to the upper core region and to the lower region of the upper plenum.

With regard to the effect of UHI on small break response, Dr. Docherty indicated that it:

1. maintains water inventory in the vessel during the transient,
2. reduces or prevents core uncovery for small break transients, and,
3. extends the range of break sizes for which no core uncovery would occur.

Dr. Docherty discussed briefly the applicability of UHI small break analysis to the WCAP-9600 study which documents the findings of a comprehensive study of small break accidents for Westinghouse's non-UHI plants. He pointed out that specific UHI phenomena were identified and then assessed to determine the applicability of UHI small break transients to the non-UHI small break studies presented in WCAP-9600. WCAP-9639 also includes specific analysis to evaluate the effect of operating the reactor coolant pumps during a small break LOCA for UHI plants.

Dr. Docherty indicated that the results of the analysis indicate that only break sizes larger than about 2 inches would result in significant UHI delivery to the reactor coolant system. Therefore, only break sizes greater than about 2 inches need be considered in the UHI plant transient response. Since break sizes 2 inches and smaller would not result in significant UHI delivery, the results of the non-UHI studies outlined in WCAP-9600 can be applied directly to UHI plants.

Dr. Docherty stated that for break sizes larger than 2 inches they have performed a UHI/non-UHI small break comparison analysis to provide a basis for assessing the applicability of UHI transients to the WCAP-9600 studies. The Westinghouse evaluation model WFLASH used in WCAP-9600 calculations was applied in the comparison analysis. The WFLASH model configuration is included in Attachment C, Page 1. The model used in the comparison analysis is somewhat different from that used in the WCAP-9600 studies since it has provision for an additional flow path from the upper head to the core. The comparison analysis was done for a 4-loop plant with typical UHI vessel internals. A typical UHI ECCS and a typical non-UHI ECCS design were analyzed. The two sets of ECCS parameters associated with each system are included in Attachment C, Page 2. Small break transient calculations were performed for break sizes larger than 2 inches. The results of the UHI/non-UHI comparison analysis are included in Attachment C, Page 3-8.

Dr. Docherty summarized the results of the UHI/non-UHI comparsion analysis:

1. The UHI system acts to add water to the primary system at high pressures, thus increasing the primary coolant inventory and preventing core uncovery at higher pressures.
2. The non-UHI transients exhibit a significant period of core uncovery at relatively high system pressure.
3. The introduction of UHI water acts to lower the primary system pressure for small break sizes where a substantial UHI flow rate is predicted.
4. A comparison of the total water inventory in the vessel for the UHI and non-UHI cases (Attachment C, Page 7) indicates that the UHI water which is delivered to the core region would remain in that region. The difference in inventory, which is shown to exist between the two cases, accounts for the majority of UHI water injected.

Dr. Docherty stated that on the basis of the results of the UHI/non-UHI comparison analysis, Westinghouse believes that the WFLASH calculated small break transients are similar in both UHI and non-UHI cases. Therefore, the description of the general behavior of the small breaks provided in WCAP-9600 would also be applicable to UHI small break transients.

Indicating that dumping a large amount of cold water through UHI system on top of a partially voided core might lead to a 3-Dimensional motion in the core, Dr. Catton asked for justification of using 1-Dimensional drift flux modeling in the small break transient calculations.

The NRC Staff pointed out that the results of some experiments performed at Dartmouth College demonstrated that any 3-Dimensional effects such as channeling would cause significantly more water to be delivered to the core. Therefore, they believe that using a 1-Dimensional model for the small break LOCA analysis is conservative.

Dr. Theofanous commented that although the results of the analysis performed by using 1-Dimensional model have enough conservatism for certain small break accidents, he does not believe that it is an adequate representation of the real situation. He does not also believe that these results would provide meaningful information to the operator to understand the real situation. Moreover, being conservative in analyzing one specific transient does not necessarily mean that the analysis for other types of transients would also be conservative.

In response to a question from Dr. Catton with regard to the type of model used by Westinghouse for calculating the froth level and the justification for using that model, Dr. Docherty stated that Westinghouse used a drift-flux model that is similar to the Zuber-Finley model. However, they have made some appropriate adjustments in the velocity term based on Westinghouse test data on core bubble rise.

In response to a question from Dr. Plesset, Dr. Docherty stated that the WFLASH model used in the analysis assumes that the water in the upper plenum will fall down very quickly into the periphery of the core.

Dr. Plesset commented that this does not seem to be a refined and improved model. He does not believe that this type of modeling will provide meaningful and adequate information to assess the accident situation.

Dr. Docherty indicated that they are in the process of doing small break accident calculations by using the advanced code NOTRUMP and they believe that they can obtain better information.

Mr. Johnson from Westinghouse commented that he believes that the UHI and non-UHI small break LOCA analyses performed by Westinghouse well predict the necessary parameters, such as pessure and temperature, that could be used by the operator to understand the system behavior under transient conditions.

Dr. Theofanous commented that the basic approach to the problem seems to be inadequate. He suggested that one better approach might be to start with a postulated initial event and then investigate the sequence of events that follow the initial event.

Dr. Catton commented that, instead of trying to justify that the small break LOCA model can handle the already existing information, it would be better to identify complete sets of variables necessary for analyzing a specific action and then determine whether the model would be able to address these variables satifactorily.

In response to a question from Dr. Schrock, Dr. Docherty stated that the superficial steam velocity is calculated at the top of the core where the active fuel starts. The steam flows up the core out into the upper plenum. Along the steam flow path there may be some changes in the cross-sectinal flow area; however, the total flow area tends to remain constant. Therefore, the steam velocity would not change significantly as the steam travels through the upper core plate and through the holes in the support columns and through the guide tubes.

In response to another question from Dr. Schrock as to how the small break LOCA calculation handles the non-equilibrium effect caused by rapid injection of cold UHI water into the steam environment, Dr. Docherty stated that the

UHI water is injected into a completely voided upper head. The rapid condensation resulting from injecting cold water into the steam space reduces the pressure in the upper head; consequently, the two phase mixture is drawn from the core into the upper head. After the upper head is filled with the initially injected UHI water, UHI delivery is terminated for some period of time. When the upper head reached the saturated stage, the saturated liquid will be drained to the upper plenum through guide tubes and support columns.

Based on the explanation given by Dr. Docherty, Dr. Schrock pointed out that, it seems that this process may be associated with some oscillatory behavior.

In response to a question from Mr. Ray as to whether Westinghouse has conducted any physical experiments to base their claim on the condensation phenomenon, Dr. Docherty stated that they have not performed any specific tests and all their claims are based on theoretical considerations. However, they have performed some sensitivity studies to determine the effect of condensation rate on the small break LOCA calculations.

With regard to Westinghouse's claim that instant depressurization of the upper head will take place upon injection of the UHI water, Dr. Theofanous commented that although cold water is injected into a steam environment the depressurizatin may not be instantaneous; it will take some time before depressurization occurs in the upper head.

In response to a question from Dr. Theofanous with regard to calculations performed by Westinghouse using an advanced model, Dr. Docherty stated that as committed to the NRC, Westinghouse has performed some advanced 1-Dimensional model calculations to study the distribution of the UHI water between the upper head and the core. This advanced model is capable of a detailed nodalization of the core. The preliminary results of this advanced model calculation indicate that the total mass distribution of water calculated by using the one node representation model is almost the same as that calculated by using the advanced model.

In response to a question from Dr. Catton as to why Westinghouse did not use some advanced multi-dimensional codes such as THINC-4 and COBRA-TF to perform more detailed small break LOCA analysis, Dr. Docherty stated that he is not sure that THINC-4 can handle counter flow phenomenon.

Dr. Catton pointed out that he was given to understand in certain previous ACRS Subcommittee meetings that THINC-4 is capable of handling counter flow. He expressed concern about the fact that Westinghouse is only trying to insure the adequacy of the 1-Dimensional model for handling 3-Dimensional questions without even trying to use some of the already existing multi-dimensional codes.

Dr. Docherty stated that one has to be careful with regard to applying multi-dimensional codes to analyze certains problems. Further, they intend to perform some scoping studies to answer some of multi-dimensional questions.

Dr. Catton expressed skepticism indicating that he does not believe that they can get answers to the questions pertinent to the multi-dimensional aspects of the flow by doing 1-Dimensional scoping calculations.

Dr. Theofanous asked about the NRC Staff's opinion on the adequacy of the approach taken by Westinghouse in analyzing the small break LOCA questions.

Dr. Plesset indicated that the NRC Staff would provide their opinion on this issue during their presentation.

In response to a question from Dr. Plesset, Mr. Johnson from Westinghouse stated that whenever the primary system pressure drops below the steam generator pressure, the steam generators will act as a heat source rather than a heat sink.

Dr. Plesset commented that, under the condition where the steam generator acts as a heat source rather than a heat sink, there will be some problem in initiating natural circulation until the system completely reverses itself. He asked whether this fact has been factored into the analysis.

Dr. Docherty stated that Westinghouse analysis does not give consideration to this fact.

In response to several questions from Dr. Schrock with regard to the energy exchange phenomenon between the steam in the upper plenum and the liquid in the upper head and the mechanics associated with the draining rate of water from the upper head through guide tubes and support columns, Dr. Docherty stated that these issues were studied extensively by Westinghouse as part of the large break UHI studies. They also conducted some semiscale and full scale (KANSAI) tests to determine the energy exchange phenomena between the steam and water and the draining rate of water through guide tubes and support columns. He pointed out that the results of these tests are documented in WCAP-8479 and also in a supplement to WCAP-8479. The NRC Staff's report on Westinghouse tests are included in NUREG-0297. He indicated that a comparison of the Westinghouse small break LOCA calculations with the KANSAI test results seems to indicate that the small break LOCA calculations under-predicts the draining rate of water.

Dr. Catton reiterated his earlier comment that Westinghouse should consider performing analysis by using the already existing 3-Dimensional codes to understand better the 3-Dimensional aspects of the UHI discharge flow.

Mr. Johnson from Westinghouse stated that they will give consideration to Dr. Catton's suggestion.

POTENTIAL IMPACT OF UHI NON-CONDENSIBLES DURING SMALL BREAK LOCA TRANSIENTS

Dr. Docherty reviewed briefly the potential impact of UHI non-condensibles during small break LOCA transients. He stated that the UHI system has potential for introducing non-condensibles into the reactor coolant system because of the use of nitrogen in the UHI system. However, in order to minimize the amount of non-condensibles introduced to the reactor coolant system during the normal UHI delivery process, the UHI system has been designed to eliminate significant water-gas surface contact area and thus minimize the absorption of nitrogen in the water. They have also established acceptance criteria to assure that less dissolved nitrogen is introduced into the reactor coolant system. The calculations performed in accordance with this criteria show that less than 45 ft^3 of dissolved gas (at Standard Temperature and Pressure (STP)) is introduced into the reactor coolant system. Actual test results has shown that less that 25 ft^3 of gas (at STP) would be introduced to the reactor coolant system through the UHI delivery process. Based on test data and analysis, they believe that significant steam generator heat removal capability is only necessary for small break LOCAs of a size less than 2 inches. Since these break sizes do not depressurize below approximately 900 psi, the volume of gas injected was evaluated at this pressure and was found to be less than 1 ft^3.

Dr. Docherty stated that based on the test data and analysis, they believe that the UHI system design specifically precludes the introduction of significant amounts of non-condensibles into the reactor coolant system. Further, the amount of dissolved gas introduced during UHI water delivery is small enough so as to be a negligible impact during the LOCA transient.

Dr. Plesset and Dr. Theofanous asked about the potential consequences of injecting all or most of the UHI nitrogen into the reactor coolant system during a small break LOCA.

Dr. Docherty responded that, based on the results of some WFLASH calculations, they believe that the potential for nitrogen injection into the reactor coolant system exists only during the small break LOCAs of a size more than 2 inches. This would also require multiple failures of the UHI isolation valves. Since there is a potential for UHI nitrogen to collect in the steam generator tubes so as to disrupt the primary to secondary heat transfer, Westinghouse analyzed the effect of UHI nitrogen on the steam generator energy removal capability. Based on the analyses, they believe that since there is no mechanism for UHI nitrogen discharge into reactor coolant system for small break LOCAs 2 inches and smaller, there should be little concern about losing steam generator heat sink due to UHI nitrogen injection. The results of the analyses also showed that steam generator energy removal is not required for break sizes (larger than 2 inches) that have potential for UHI nitrogen injection because during such transients the decay heat can be removed through the break.

Dr. Docherty stated that they have also analyzed the effect of UHI nitrogen on the energy removal capability of breaks. Based on the analysis, they believe that the UHI nitrogen might replace about one percent of the steam flow through the break and hence the energy removal capability of the system through the break would be about one percent less. They have also analyzed a case assuming complete discharge of nitrogen (but no steam) through the break. Under this situation, since the steam energy is not removed through the break, the system pressure would go up. However, since all the nitrogen would be discharged through the break within a very short period of time,the system has very little time for repressurization.

Dr. Plesset asked whether they have performed any analysis to determine the consequences assuming that all UHI nitrogen is discharged into the reactor coolant system and the operator isolates the break by closing appropriate valves.

Dr. Docherty responded that based on the results of the Westinghouse's calculations, they believe that there would be about 500 ft^3 of nitrogen in the system; the total steam generator tube volumes would be about 3100 ft^3. Although there is a potential for nitrogen to deposit on the steam generator tubes so as to disrupt the primary to secondary heat transfer, it is not likely that the nitrogen would degrade all the steam generator tubes. Therefore, they believe that adequate steam generator heat transfer surface remains available even if the entire UHI nitrogen is discharged into the reactor coolant system.

Dr. Plesset asked what the operator would do under the circumstance where all UHI nitrogen is discharged into the reactor coolant system and the break is isolated subsequent to that nitrogen discharge. He also asked whether the expected operator reaction has been factored into the Westinghouse's calculations.

Dr. Docherty stated that since it is hard to predict the expected operator reaction, it has not been factored into the calculations.

Dr. Plesset stated that it would be better to give consideration to operator reaction under such unusual situations.

In response to a question from Dr. Zudans as to what would be the consequences if the UHI is not initiated at the time it is supposed to be and it is initiated at the time it is not supposed to be, Dr. Docherty stated that they have not performed any specific analysis to determine the consequences of this situation. However, based on the evaluation model calculations, they believe that if the UHI fails to deliver water during a small break or large break transients the peak clad temperature would go up slightly. But, still they believe that the system would be able to survive under these circumstances.

Dr. Theofanous commented that although the system may be able to survive under the circumstance discussed above, he does not want to have a situation where the UHI does not deliver water when it is supposed to be.

The presentation material pertinent to the evaluation UHI nitrogen discharge into reactor coolant system is included in Attachment E.

In response to a question from Dr. Zudans as to whether Westinghouse has looked at the effects of UHI injection on certain components located in the upper head region, Dr. Docherty stated that they have looked at the effects of UHI injection on the functions of various components such as guide tubes, and support columns; based on the review, they believe that UHI injection would not cause loss of function of any of these components.

PRESENTATION BY THE NRC STAFF

Prior to the NRC Staff's scheduled presentations, Mr. Phillips provided some comments with regard to some of the earlier concerns expressed by the Subcommittee and its consultants.

With regard to the NRC Staff's short-term program for improving operator guidelines, he stated that the vendors had submitted the inadequate core cooling analysis for NRC Staff's review and that is being reviewed by the NRC Staff. The best-estimate analyses pertinent to transients and accidents are expected to be submitted to the NRC Staff in the very near future. The NRC Staff believes that the code used by one of the vendors for best-estimate analysis is somewhat different from that normally used in the safety analysis.

Mr. Phillips stated that most of the vendors use evaluation models in the large break LOCA analysis except one vendor who claims to have a best-estimate code for large break analysis. Based on the review of the available information, the NRC Staff believes that an evaluation model, with best-estimate inputs, produces as good results as that of a best-estimate code.

With regard to the models used in the small break LOCA analyses, Mr. Phillips pointed out that, the NRC Staff believes that they are adequate in certain areas. They have identified several shortcomings in these models and the vendors have been asked to refine their models so as to take care of these shortcomings. The NRC Staff intends to review the results of the refined models. The NRC Staff also intends to audit vendors' calculations by using some of the NRC Staff's codes. They will limit the application of some of the vendors' codes if they find that these codes will not be able to handle or predict certain issues such as the effects of the termination of natural circulation. With regard to the adequacy of using 1-Dimensional codes to analyze 3-Dimensional effects, the NRC Staff, based on the past experience, believes that 1-Dime--ional codes are adequate for most of these calculations.

Dr. Catton commented that he still believes that some analysis should be performed using 3-Dimensional codes to get appropriate answers to the 3-Dimensional aspects of the problem.

In response to a question from Dr. Catton as to whether there is any plan to perform an analysis to determine the relative consequences of pump "on" when it should be "off" and "off" when it should be "on", Mr. Phillips stated that the NRC Staff has asked the vendors to identify the types of analyses expected to be performed and submit that information to the NRC for review; however, the vendors have not been specifically directed to analyze the pump "on" and "off" issue in their short-term programs.

Dr. Plesset stated that the ACRS would like to review the vendors' submittals on small break LOCAs as they become available.

In response to a question from Dr. Theofanous with regard to the NRC Staff's position on the adequacy of the vendors' models, Mr. Phillips stated that, based on their review, the NRC Staff has concluded that the vendors' models are adequate in predicting the system behavior to a certain extent so as to provide guidelines to the operator to handle emergency situations. However, the NRC Staff has also identified several shortcomings in the vendors' models and instructed the vendors that their model should be improved to preclude the deficiencies identified by the NRC Staff.

Dr. Theofanous wondered how the NRC Staff could conclude that the models are adequate but at the same time identify deficiencies. He believes that the basic approach to the problem seems to be inadequate. Even if the vendors make changes to their models in accordance with the directions of the NRC Staff, he does not believe that the modified models would provide adequate answers to all of the unresolved issues. As far as he is concerned, the basic problem lies with the input to the models; the input to the models are inadequate. He believes that some of the other inputs should be explored and factored into the calculations and also some of the advanced multi-dimensional codes should be used to get satisfactory answers to the unresolved issues.

Mr. Phillips pointed out that the NRC Staff has a contract with the Los Alamos Laboratory to perform small break LOCA calculations using the TRAC code. They also intend to use the TRAC model to audit the vendors' calculations.

Mr. Johnson from Westinghouse stated that they are pursuing the possibility of using several advanced multi-dimensional codes such as TRAC and COBRA-II in the small break LOCA calculations. However, based on the results of the calculations already performed, Westinghouse strongly believes that 1-Dimensional codes are adequate to predict the overall system behavior so as to provide guidelines to the operators to handle emergency situations.

Dr. Theofanous commented that he does not believe that there is adequate experimental basis to support Westinghouse's claim.

Mr. Johnson stated that there have been some small break tests performed at INEL at the Semi-Scale and LOFT Facilities and the system behavior has been well predicted in those tests.

Dr. Zudans provided his opinion indicating that he believes that the overall system behavior can be well predicted by the models used by Westinghouse in the small break LOCA calculations.

NRC STAFF'S REVIEW OF WESTINGHOUSE UHI SMALL BREAK LOCA CALCULATIONS (WCAP-9639) - MR. HOLAHAN

Mr. Holahan stated that the main objectives of the NRC Staff's review of Westinghouse UHI small break LOCA calculations are to:

1. Assure effective emergency procedures for handling small break accidents.

2. Identify weaknesses, if any, in the ECCS evaluation model and make recommendations for licensing action.

Mr. Holahan pointed out that WCAP-9639 report was issued by Westinghouse in December 1979; it was reviewed by the NRC Staff and the questions that arose as a result of that review were sent to Westinghouse in February 1980; Westinghouse's response to these questions, received in March 1980, are being reviewed by the NRC Staff. The NRC Staff intends to complete the review of WCAP-9639 in April 1980.

The basic approach used in the small break LOCA calculations were discussed.

Dr. Theofanous reiterated one of his earlier comments indicating that one has to first understand the behavior of the system under a given set of boundary conditions and then decide what actions should be or should not be taken by the operator. Based on the information presented to the Subcommittee, he believes that this approach has not been followed in the small break LOCA calculations. He expressed concern about the fact that either Westinghouse or the NRC Staff failed to indicate that the small break LOCA calculation done so far is a stepping stone towards a more detailed and comprehensive study.

Mr. Holahan sta' .d that they do have a long-term program to perform additional analyses on th's issue, but he is not certain whether that will answer all of the concerns expressed by Dr. Theofanous.

Mr. Holahan summarized the NRC Staff's findings on the UHI plant behavior during small break LOCAs:

1. Small break LOCAs of a size less than 2 inches do not result in significant UHI delivery.

2. For small break LOCAs of sizes 6 i..ches and larger, core recovery and peak clad temperature are controlled by the cold leg accumulator injection; UHI water is delivered during the early part of the transient thus delaying the core uncovery; however, it does not prevent eventual core uncovery.

3. The effect of UHI delivery during small break LOCA transients of sizes in the range of 3 inches to 6 inches seems to be very beneficial. For these types of small break LOCAs the UHI delivery prevents a prolonged core uncovery which would have occurred without UHI.

In response to a question from Mr. Ray with regard to the possibility of increasing the accumulator volume so as to increase the quantity of water during UHI delivery, Mr. Docherty stated that dumping too much water other than necessary does not necessarily provide adequate core cooling; further, excessive water may cause back pressure in the system.

In response to a question from Dr. Plesset as to what prevents the water in the upper head from draining instantaneously, it was pointed out that the pressure difference between the upper plenum and the upper head draws the steam towards the upper head and the steam which flows toward the upper head tends to provide some support to the water in the upper head thus preventing instantaneous drainage.

Status of Outstanding Issues

Mr. Holahan reviewed briefly the status of several of the outstanding issues:

Issues That Have Been Resolved

Mr. Holahan stated that based on the review of WCAP-9639, the NRC Staff believes that the following issues have been resolved:

1. Small break emergency procedure guidelines.

   For this issue, the guidelines applicable for non-UHI plants will also be applicable to UHI plants. In addition, Westinghouse proposed to include guidelines to the operator to verify the closure of the UHI isolation valves when the UHI accumulator water reaches low level; if the isolation valves haven't been closed by the low-level signal, the operator has to close them manually.

2. Analysis of UHI Line Break.

Issues Near Resolution

1. Influence of UHI on the symptoms of inadequate core cooling.

   Westinghouse attempted to resolve this item at least for 5% power conditions by providing calculations to show that UHI delivery would not mislead the oper or with regard to the actual core conditions.

2. Probability and consequences of failure of the UHI isolation system.

3. Modeling of UHI non-equilibrium conditions in the upper head.

4. Influences of the upper head thermal equilibrium model on the small break accident.

5. Small break model nodalization.

Issues That Are Open

1. Influences of UHI on natural circulation.

   So far Westinghouse has performed some preliminary calculations on this issue. The results of these calculations seem to indicate that the core temperatures would remain at saturation during the whole period of time until the UHI water is heated enough to reinstate natural circulation.

2. Modeling of upper plenum.

   Under this issue there are some questions about the adequacy of the single node representation of the upper plenum and the core. There are also some questions with regard to the manner in which water drains from the upper plenum into the core. Westinghouse tries to justify the adequacy of the model (single node representation of the core and upper plenum) by performing calculations using NOTRUMP code; if not justified, NRC Staff may require that Westinghouse should use more detailed nodalization for the core.

Mr. Holahan provided a brief summary indicating that the NRC Staff requires that:

1. all the unresolved issues should be resolved before 5% power testing at the Sequoyah plant,

2. additional study of ECCS evaluation model for small break accidents in UHI plant is needed (Attachment E).

In response to questions from Dr. Theofanous and Dr. Catton with regard to the reasons for not using the NOTRUMP code instead of the WFLASH code in the small break LOCA calculations, Mr. Johnson stated that NOTRUMP code does not have as much conservatism as the WFLASH code.

Indicating that he was under the impression that NOTRUMP code is better than the WFLASH code, Dr. Catton wondered as to why the NRC Staff does not require that NOTRUMP code should be used in the small break LOCA calculations.

Mr. Lauben from the NRC Staff stated that they will look into this issue and if found necessary, they may require the use of NOTRUMP code in the small break LOCA calculations.

The Subcommittee suggested tht Westinghouse should provide a comparison of the results of the calculations done by using NOTRUMP and WFLASH codes.

ECCS CAPABILITY FOR OLDER, CURRENT AND FUTURE PLANTS - MR. LAUBEN

Mr. Lauben pointed out that the generic issues associated with the ECCS capability were considered resolved as a result of the promulgation of ECCS regulation outlined in 10 CFR 50.46 and Appendix K. Even after the promulgation of the ECCS regulation, the ACRS has repeatedly pointed out that the NRC Staff should seek some improvements in the ECCS capability for future plants. Subsequent to the Three Mile Island, Unit-2 (TMI-2) accident, the TMI-2 Task Action Group identified several issues for improving the ECCS capability (Attachment F, Pages 1 and 2) The TMI-2 Task Action Plan recommends that these issues should be resolved and consideration should be given to modify, as appropriate, existing ECCS models, Standard Review Plan and 10 CFR 50.46 and Appendix K. The NRC Staff plans to evaluate the adequacy of the existing ECCS models by obtaining necessary information from the vendors and licensees; based on their evaluation, they may make appropriate changes to the ECCS models, Standard Review Plan and 10 CFR 50.46 and Appendix K.

With regard to the ECCS capability of future plants, Mr. Lauben pointed out that the NRC Staff does not intend to treat the future plants differently from older or current plants. They believe that any new ECCS requirements resulting from the TMI-2 accident would be equally applicable to all plants.

Mr. Ray commented that he believes that the TMI-2 accident did not pinpoint any specific deficiencies in the existing ECCS or in the ones that have been promulgated for future plants. He also believes that several issues such as the need for better understanding of the ECCS and improved operating procedures were the ones identified as a result of the TMI-2 accident.

Mr. Lauben stated that he agrees with Mr. Ray's comment. He stated further that the NRC Staff has not yet decided whether to make changes to the ECCS regulation outlined in 10 CFR 50.46 and Appendix K as a result of the TMI-2 accident.

REACTOR COOLANT PUMP OVERSPEED DURING A LOCA

Mr. Ignatonis indicated that General Electric (GE) Company had performed a study in 1972 on the reactor coolant pump overspeed issue. Based on the preliminary results of that study, GE initially thought that use of decouplers may provide pump overspeed protection. However, based on the results of the revised analysis performed in 1978, GE believes that decouplers are not needed for pump overspeed protection. He stated that GE's analysis and conclusions have been documented and submitted to the NRC Staff for review and the NRC Staff has not yet completed its review. The NRC Staff has sought some technical assistance from the Shaker Research Corporation to determine the adequacy of GE conclusions. A comparison of GE conclusions and Shaker Research Corporation conclusions is included in Attachment F, Page. 3.

Mr. Ignatonis reviewed briefly the analysis performed by the NRC Staff using the RELAP-4/MOD-5 code (Attachment F, Pages 4 and 5) and some of the analysis performed by B&W using the CRAFT code (Attachment F, Page 6). He also discussed briefly some of the methods that are being considered for reducing reactor coolant pump overspeed (Attachment F, Page 7).

After the discussion of all the scheduled items, Dr. Plesset requested that the ACRS consultants who were present at the subject meeting provide their opinions in writing on the matters discussed.

Dr. Plesset thanked all the participants and adjourned the meeting at 4:45 p.m.

****

NOTE: For additional details, a complete transcript of the meeting is available in the NRC Public Document Room, 1717 H St., NW, Washington, DC 20555 or from International Verbatim Reporters, Inc., 499 South Capitol Street, SW, Suite 107, Washington, DC 20002, (202) 484-3550.

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Subcommittee on Emergency Core Cooling Systems; Notice of Meeting

The ACRS Subcommittee on Emergency Core Cooling Systems will hold a meeting on March 26, 1980 in Room 1046, 1717 H St., NW, Washington, DC 20555.

In accordance with the procedures outlined in the Federal Register on October 1, 1979 (44 FR 56408), oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

The agenda for subject meeting shall be as follows:

Wednesday, March 26, 1980: 8:30 a.m. Until the Conclusion of Business Each Day

The Subcommittee may meet in Executive Session, with any of its consultants who may be present, to explore and exchange their preliminary opinions regarding matters which should be considered during the meeting.

At the conclusion of the Executive Session, the Subcommittee will hear presentations by and hold discussions with representatives of the NRC Staff, Westinghouse, and other interested persons regarding the analysis of small break LOCAs in Westinghouse UHI reactors. The Subcommittee will also review several ACRS generic items related to the capability of ECCS systems.

In addition, it may be necessary for the Subcommittee to hold one or more closed sessions for the purpose of exploring matters involving proprietary information. I have determined, in accordance with Subsection 10(d) of the Federal Advisory Committee Act (Public Law 92-463), that, should such sessions be required, it is necessary to close these sessions to protect proprietary information. See 5 U.S.C. 552b(c)(4).

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant Designated Federal Employee, Dr. Andrew L. Bates (telephone 202/634-3267) between 8:15 a.m. and 5:00 p.m., EST.

Dated: March 5, 1980.

John C. Hoyle,

Advisory Committee, Management Officer.

[FR Doc. 80- 7?8 Filed 3-10-80; 8:45 am]

BILLING CODE 7590-01-M

ATTACHMENT B

ACRS SUBCOMMITTEE MEETING ON
EMERGENCY CORE COOLING SYSTEMS
MARCH 25, 1980
WASHINGTON, DC.

ATTENDEES LIST

ACRS

M. Plesset, Chairman
J. Ray, Member
H. Etherington, Member
I. Catton, Consultant
V. Schrock, Consultant
J. Lienhard, Consultant
T. Theofanous, Consultant
Z. Zudans, Consultant
A. Bates, Staff*
S. Duraiswamy, Staff

*Designated Federal Employee

WESTINGHOUSE

W. Johnson
P. Docherty

NRC

R. Audette, DSS/AB
H. Woods, IE
L. Phillips, DSS/AB
N. Lauben, NRR
G. Holahan, NRR
G. Mazetis, DDS/RSB
A. Ignatonis, NRR

McGRAW HILL

J. Dann

IVRI - REPORTER

V. Smith
S. Corsanico
N. Pilpaolo

ATTACHMENT B

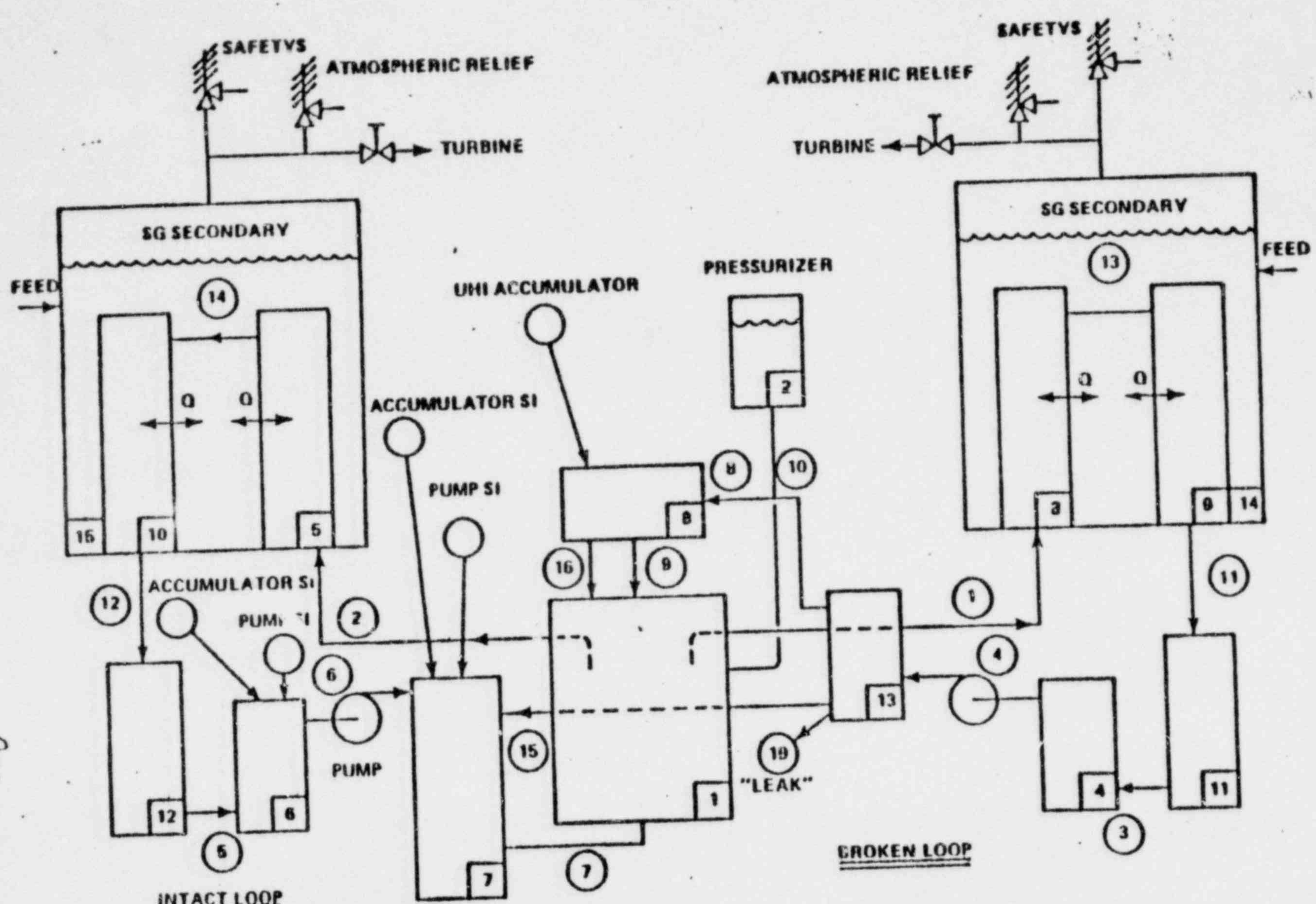


Figure 1 WFLASH Model for a Westinghouse PWR Upper Head Injection

15802-10

ATTACHMENT C

REFERENCE PLANT COMPARISON

UHI SYSTEM PARAMETERS

UHI ACCUMULATOR

PRESSURE	1300 PSIA
WATER VOLUME	710 FT^3
FL/D	23

COLD LEG ACCUMULATOR

PRESSURE	400 PSIA
WATER VOLUME	1050 FT^3 PER ACCUMULATOR
FL/D	23

NON UHI SYSTEM

COLD LEG ACCUMULATOR

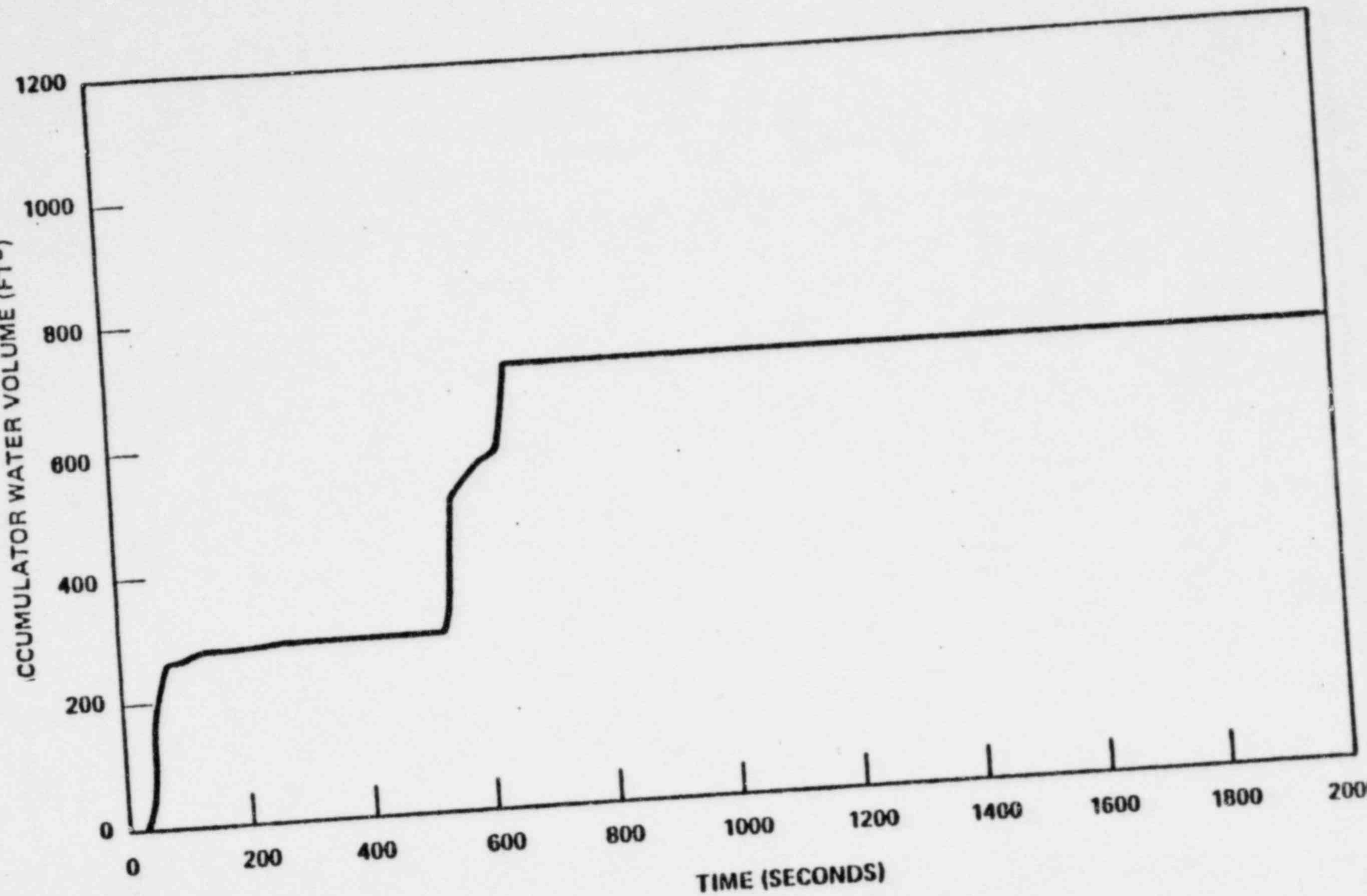
PRESSURE	600 PSIA
WATER VOLUME	1000 FT^3 PER ACCUMULATOR
FL/D	7.36

UHI/NON UHI COMPARISON

ANALYSIS RESULTS

CASE	CORE UNCOVERY TIME	PEAK CLAD TEMPERATURE RESULTS
8 INCH UHI	138. SECONDS	1083
8 INCH NON-UHI	238. SECONDS	835
6 INCH UHI	580 SECONDS	1528
6 INCH NON UHI	416 SECONDS	1464
4 INCH UHI	NO SIGNIFICANT UNCOVERY	
4 INCH NON-UHI	1050 SECONDS	1690
3 INCH NON-UHI	1420 SECONDS	1243

C-4



15802-4

Figure 26 4 inch

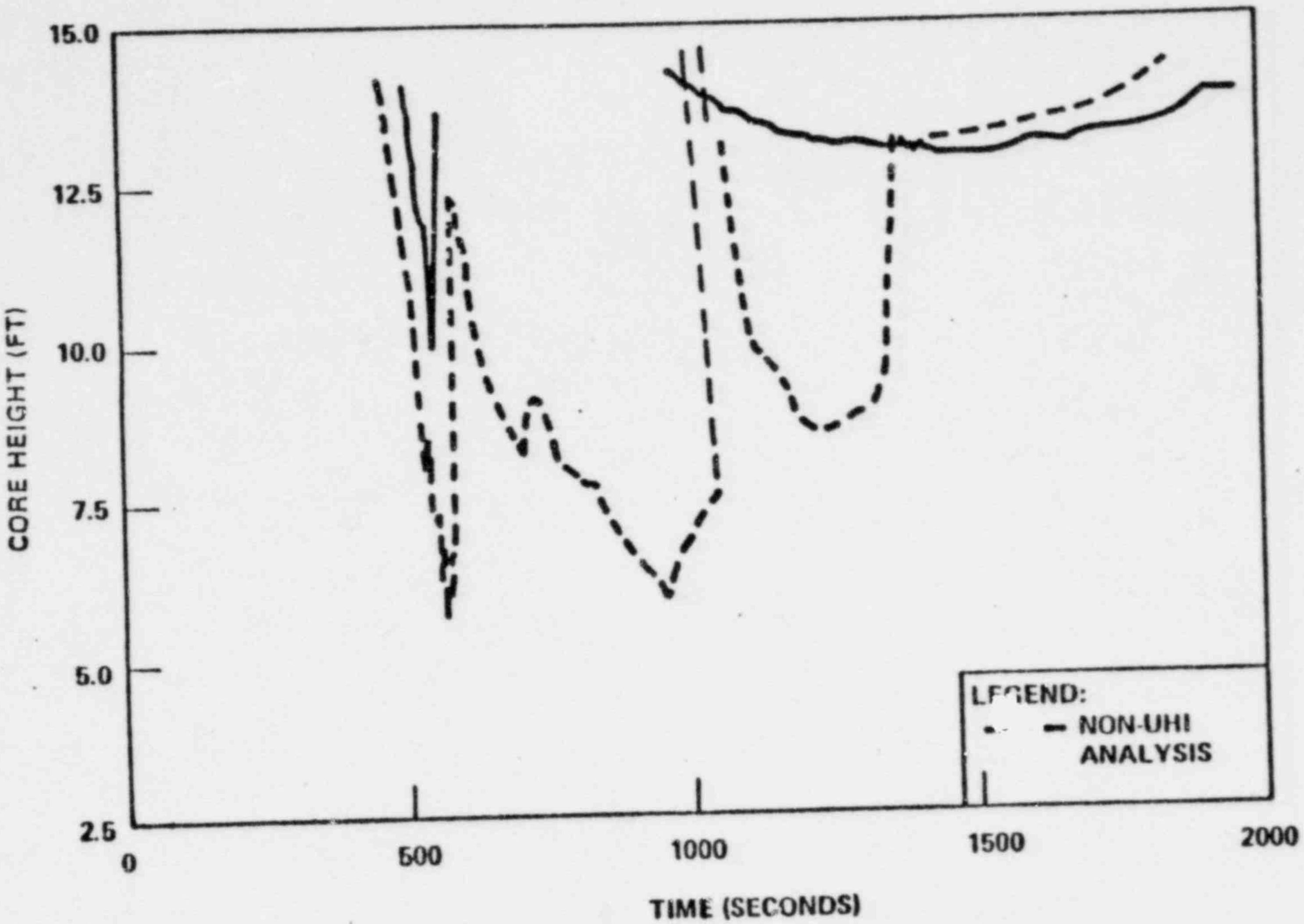


Figure 27 4 Inch Cold Leg Break Core Height (UHI/Non UHI) Comparison

15802-6

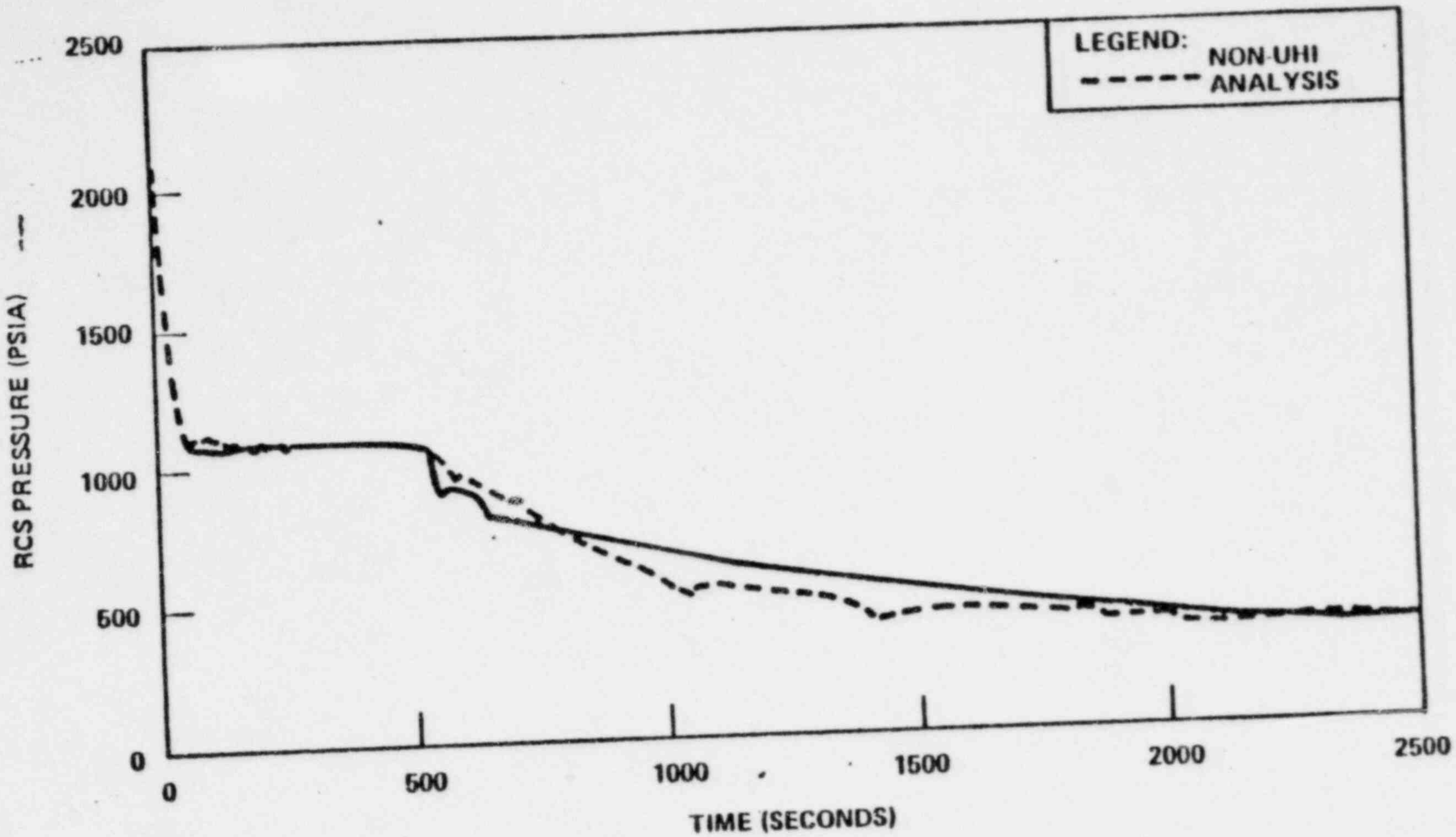
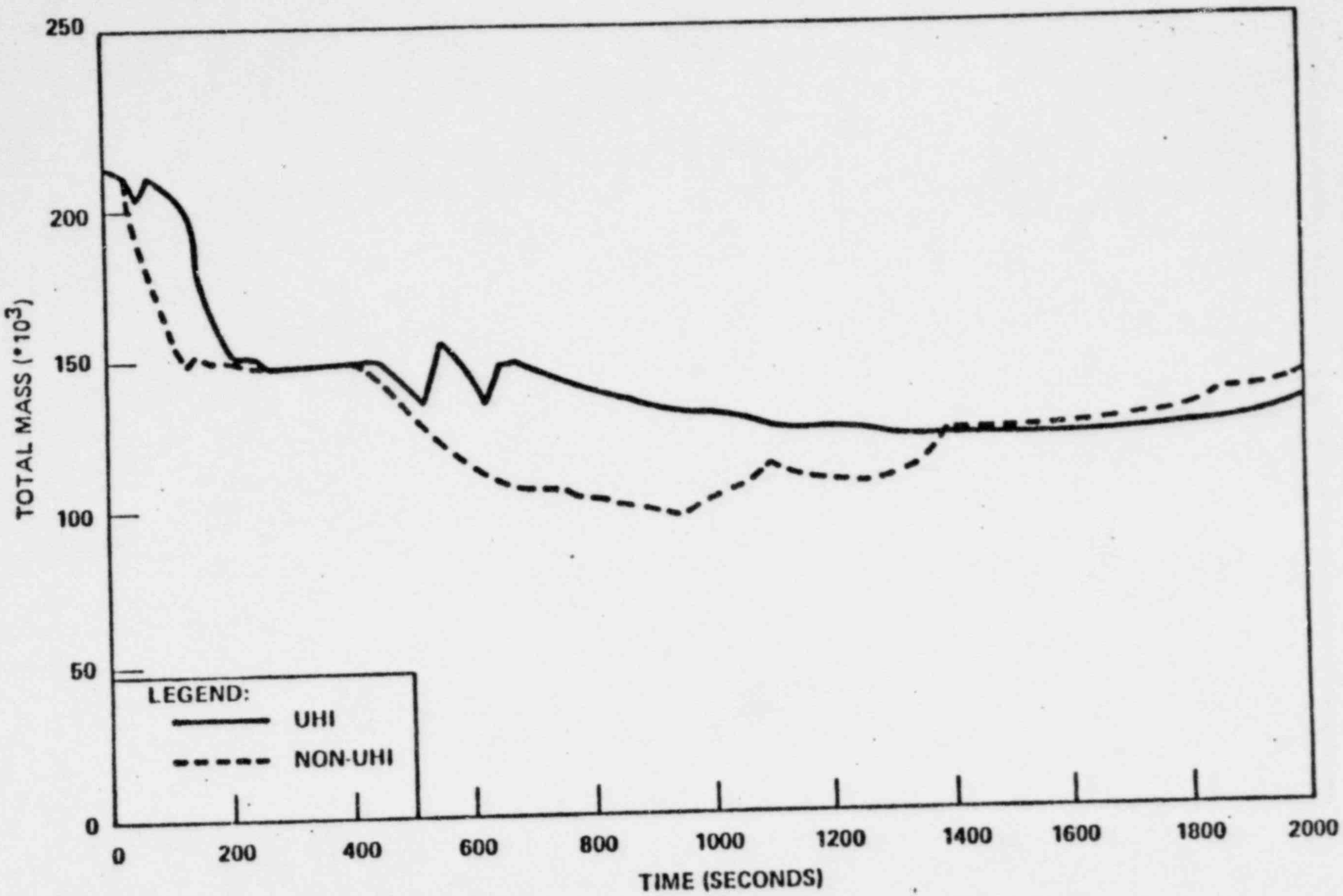


Figure 28 4 Inch Cold Leg Break RCS Pressure (UHI/Non UHI) Comparison

15802-7



15802-5

Figure 29 4 Inch Break

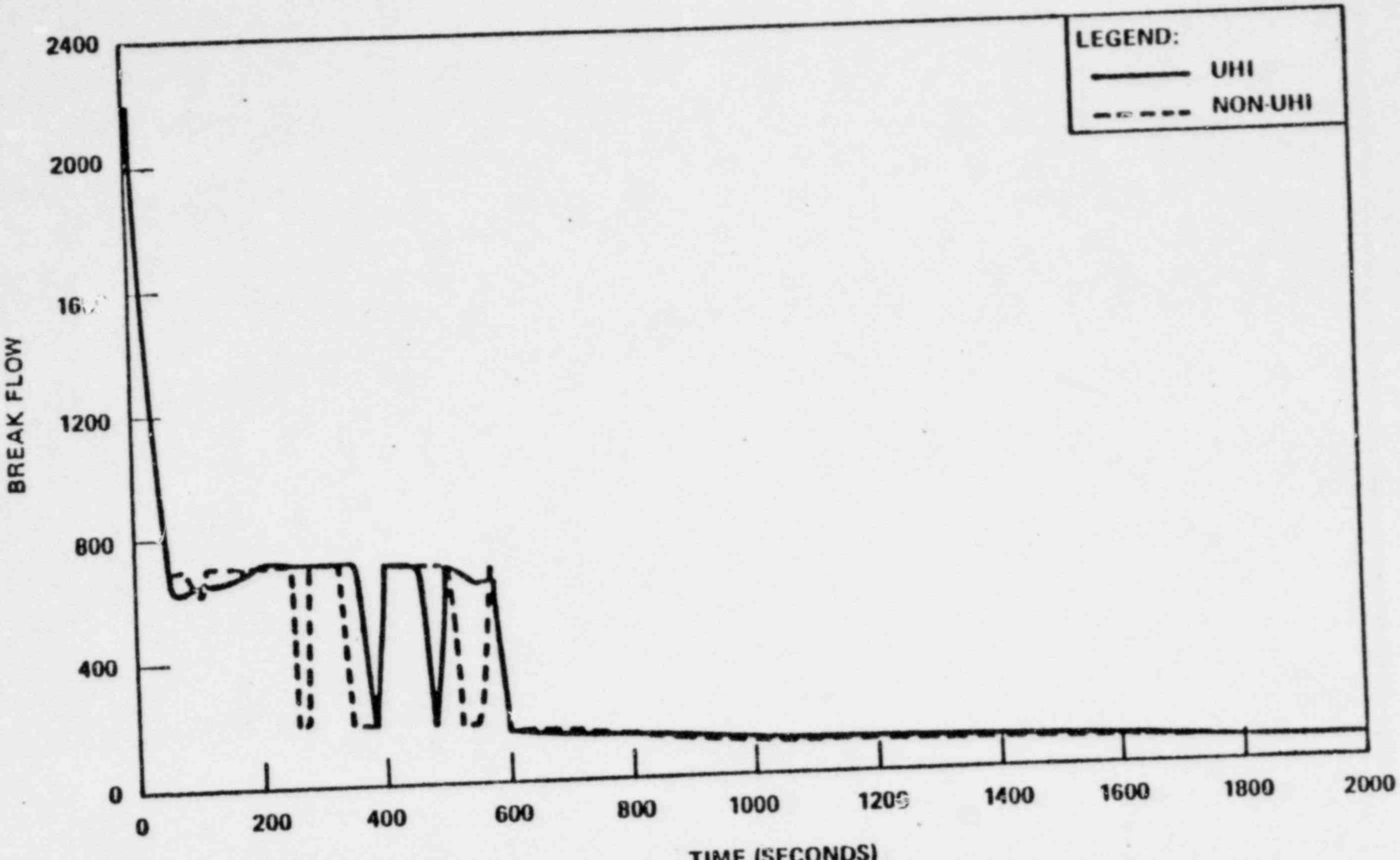


Figure 30 4 Inch

EFFECT OF UHI NITROGEN ON STEAM GENERATOR ENERGY REMOVAL

POTENTIAL FOR UHI NITROGEN TO COLLECT IN THE STEAM GENERATOR TUBES DISRUPTING PRIMARY TO SECONDARY HEAT TRANSFER

EXAMINE RELATIONSHIP BETWEEN SMALL BREAK TRANSIENTS WHICH REQUIRE STEAM GENERATOR ENERGY REMOVAL

SMALL BREAK TRANSIENTS WHICH HAVE POTENTIAL FOR UHI NITROGEN INJECTION

STEAM GENERATOR ENERGY REMOVAL IS NOT REQUIRED FOR BREAK SIZES THAT HAVE POTENTIAL FOR UHI NITROGEN INJECTION

ATTACHMENT D

EFFECT OF UHI NITROGEN ON BREAK ENERGY REMOVAL

POTENTIAL FOR UHI NITROGEN DISCHARGE AT THE BREAK

NITROGEN DISCHARGE WOULD REPLACE A PERCENTAGE OF THE STEAM BREAK FLOW

CONSEQUENCES

PRIMARY SYSTEM REPRESSURIZATION

POTENTIAL PRIMARY FLUID INVENTORY LOSS FROM SI FLOW/BREAK STEAM FLOW MISMATCH

PRIMARY SYSTEM REPRESSURIZATION

POTENTIAL FOR PRIMARY SYSTEM OVERPRESSURIZATION ASSESSED VIA A BOUNDING CALCULATION

CALCULATION ASSUMPTIONS

APPLIED TO THE SMALL BREAK TRANSIENT WITH THE MAXIMUM VOLUME OF UHI NITROGEN (8 INCH BREAK)

APPLIED FOR 100% NITROGEN DISCHARGE

ASSUMES CONSTANT PRIMARY SYSTEM PRESSURE OF 120 PSIA

RESULT:

LESS THAN 90 SECONDS REQUIRED TO DISCHARGE ALL INJECTED NITROGEN

PRIMARY SYSTEM OVERPRESSURIZATION IS NOT EXPECTED TO OCCUR

STEAM GENERATOR HEAT REMOVAL WILL BE RESTORED AS PRIMARY SYSTEM PRESSURE INCREASES

FIGURE 7


RCS
PRESSURE
(PSIA)
2000
1600
1200
800
400
0
20
40
60
80
100
% BREAK AREA TAKEN UP
BY N_2 DISCHARGE

PRIMARY SYSTEM REPRESSURIZATION

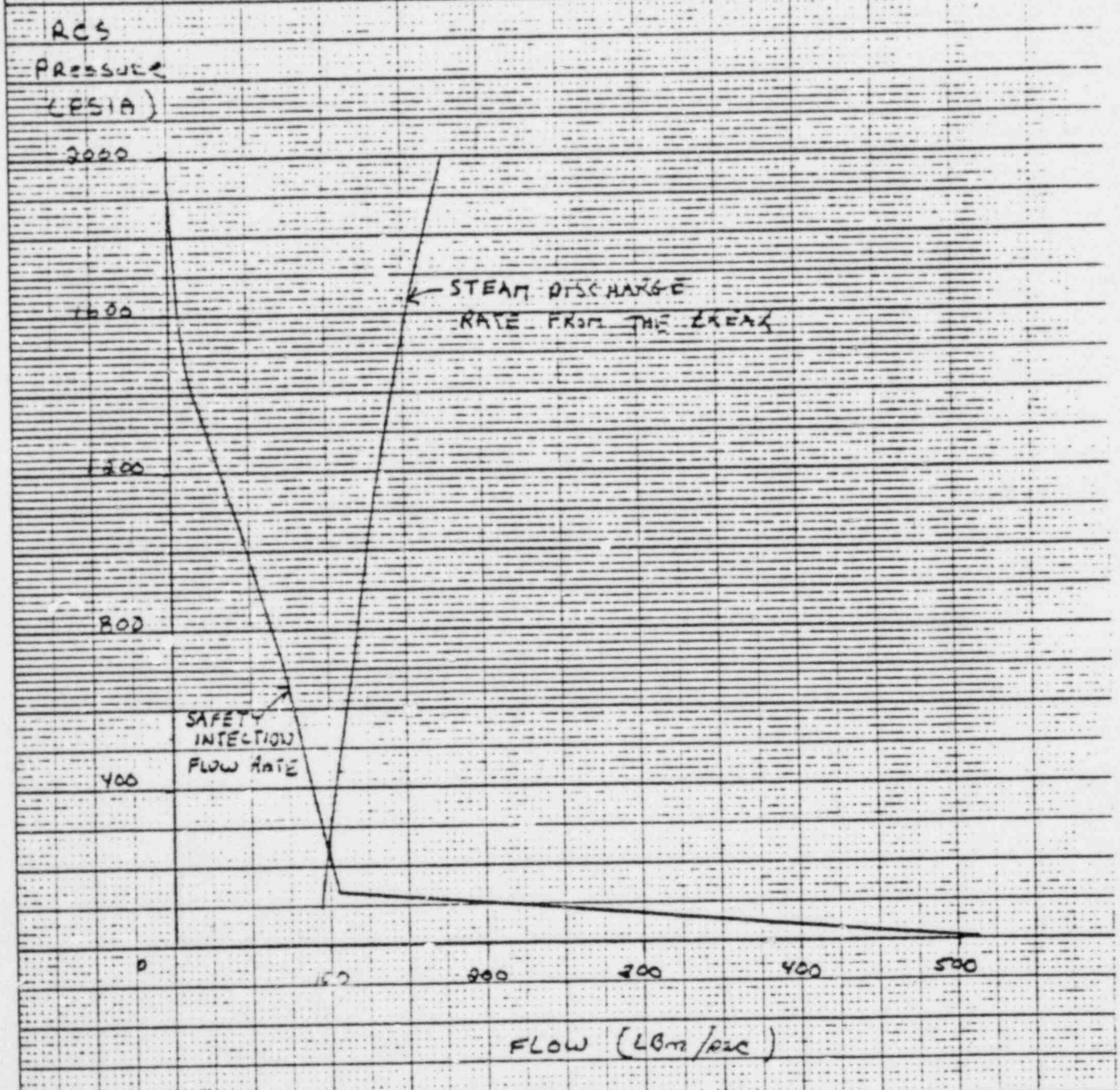
POTENTIAL FOR PRIMARY SYSTEM FLUID INVENTORY LOSS DURING A PARTIAL NITROGEN DISCHARGE CONDITION

HYPOTHESIZE A DISCHARGE CONDITION THAT RESULTS IN A REPRESSURIZATION TO A PRIMARY SYSTEM PRESSURE AT WHICH

$\dot{m}_{SI}$ IS LESS THAN $\dot{m}_{BREAK}$

INVENTORY LOSS IS THE RATE MISSMATCH TIMES THE INTERVAL FOR NITROGEN DISCHARGE

Figure 8



RCS
PRESSURE
(PSIA)
2000
1600
1200
800
400
0
STEAM DISCHARGE RATE FROM THE BREAK
SAFETY INJECTION FLOW RATE
200
500
FLOW (LBm/SEC)

Table 3

POTENTIAL INVENTORY LOST DURING

N_2 DISCHARGE

PRIMARY SYSTEM PRESSURE PSIA	PERCENTAGE BREAK AREA TAKEN UP BY N_2 FLOW	TIME TO PURGE N_2 (SECONDS)	LOSS RATE LBS/SEC	TOTAL LOST LBS
2000	88.6%	50.0	175.	8750
1600	86.4	52.2	139.	7255.
1200	84.6	62.0	85.	5270.
800	79.3	68.5	44.6	3055.
400	63.3	77.2	10.0	772.
200	33.0	84.4	0.0	0.0

EFFECT OF UHI NITROGEN ON MAXIMUM CONTAINMENT PRESSURE

THE INCREMENTAL PRESSURE INCREASE FROM UHI NITROGEN DISCHARGE TO CONTAINMENT IS ESTIMATED BY RELATING MASS OF UHI NITROGEN TO TOTAL AIR MASS IN THE CONTAINMENT

FOR TYPICAL UHI PLANT CONTAINMENT UHI NITROGEN DISCHARGE WOULD RESULT IN 13 PERCENT INCREASE OF MASS OF NON-CONDENSIBLES

PRESSURE INCREASE FROM UHI NITROGEN IS EXPECTED TO BE IN THE RANGE OF 13 PERCENT

CONSEQUENCES OF UHI VALVE FAILURE

VALVE OPERATION IS REQUIRED ONLY FOR SMALL BREAK LOCA TRANSIENT OF 2 INCH DIAMETER AND LARGER BREAK SIZES

NO SEVERE CONSEQUENCES HAVE BEEN IDENTIFIED AS THE RESULT OF UHI NITROGEN INJECTION DURING SMALL BREAK TRANSIENTS

SUMMARY & CONCLUSIONS

SMALL BREAK ANALYSIS, GUIDELINES AND PROCEDURES

1. SMALL BREAK ACCIDENT ANALYSES HAVE BEEN PERFORMED FOR UHI PLANTS.

2. TWO (2) REVIEW AREAS ARE RESOLVED.

3. FIVE (5) REVIEW AREAS ARE NEAR RESOLUTION.

4. TWO (2) REVIEW AREAS ARE OPEN.

CONCLUSIONS

1. ALL REVIEW ITEMS MUST BE RESOLVED BEFORE 5% POWER TESTING.

2. ADDITIONAL STUDY OF ECCS EVALUATION MODEL FOR SMALL BREAK ACCIDENTS IN UHI PLANTS IS NEEDED.

    A. THERMAL EQUILIBRIUM MODEL IN UPPER HEAD.

    B. UPPER PLENUM AND HOT LEG NODALIZATION.

    C. THERMAL EQUILIBRIUM IN COLD LEG INJECTION LOCATIONS.

ATTACHMENT E

TASK II.E.2 ECCS

OBJECTIVES

1. DECREASE RELIANCE ON ECCS FOR NON-LOCA EVENTS.

2. ENSURE THAT ECCS DESIGN-BASIS RELIABILITY AND PERFORMANCE ARE CONSISTENT WITH OPERATIONAL EXPERIENCE.

3. ACHIEVE BETTER UNDERSTANDING OF ECCS PERFORMANCE.

4. ENSURE PROPER TREATMENT OF SB LOCA PREDICTION UNCERTAINTIES.

ACTION ITEMS

1. SURVEY LICENSEES AND APPLICANTS ON ECCS ACTUATION EXPERIENCE. REVIEW AND REQUIRE PROCEDURE OR TECH. SPEC. CHANGES AS REQUIRED.

2. RE-FOCUS RESEARCH ON SB LOCA AND ANOMALOUS TRANSIENTS.

3. REVIEW UNCERTAINTIES IN SB LOCA ANALYSIS. CONSIDER CHANGES TO ECCS MODELS, SRP, ECCS RULE (10 CFR 50.46 AND APPENDIX K).

ATTACHMENT F

TASK II.K. MEASURES TO MITIGATE SMALL-BREAK LOCA AND LOFW EVENTS

OBJECTIVES:

PERFORM SYSTEM RELIABILITY ANALYSES AND EFFECT CHANGES IN EMERGENCY OPERATING PROCEDURES TO IMPROVE CAPABILITY TO MITIGATE SB LOCA AND LOFW EVENTS.

ACTION ITEMS:

1. COMPLETE EVALUATIONS OF RESPONSES TO I&E BULLETINS.
2. O.L. APPLICANTS TO ADDRESS SAME ISSUE.
3. NRR & SD CODIFY REQUIREMENTS.
4. IMPLEMENT B&O FINAL RECOMMENDATIONS RELATED TO:
   - A. IMPROVED EMERGENCY PROCEDURES.
   - B. INPROVED OPERATOR TRAINING.
   - C. SYSTEM MODIFICATIONS.
   - D. ADDITIONAL ANALYSIS.

ATTACHMENT F

TECH ASSIST FOR GE REVIEW

KEY ITEM	CONCLUSIONS GE	SRC
SUDDEN PUMP SEIZURE; RESULT IN MISSILE	VERY UNLIKELY (ANALYZED)	AGREE
MAX/MIN TORQ. FOR SHAFT FAILURE	$5.6 \times T_R / 3.45 \times T_R$	$8.0 \times T_R / 3.2 \times T_R$
ROTOR-TO-ROTOR GAP CLOSURE	$2.92 \times N_R$	$4.2 \times N_R$

NRC INDEPENDENT PUMP SPEED ANALYSIS

- UTILIZED RELAP 4/MOD 5 CODE
- B&W, OCONEE UNITS 1 & 2 (FA-177 CLASS)
- COMPARED 1) 2Ø PUMP HEAD & TORQUE MULTIPLIERS (C-E/EPRI vs. ANC SEMI-SCALE)
  2) LICENSING vs. BEST ESTIMATE BREAK FLOW
- RESULT: 1) NO CHANGE IN MAX. PUMP SPEED
  2) 366% x N_R (LICENSING)

  vs.

  272% x N_R (BEST ESIMATE)

F4

SUMMARY

- 2Ø PUMP HEAD & TORQUE MULTIPLIERS DO NOT SIGNIFICANTLY CHANGE ORIGINAL VENDOR CALCS.
- ANTICIPATE NEW ORGANIZATION WILL CONTINUE THIS REVIEW.

B&W PUMP SPEED CALCULATIONS

USING CRAFT CODE

PLANT SIZE	177-FA	205-FA
BREAK SIZE	8.55 FT^2 DOUBLE-ENDED →	
PUMP MODEL	SHOWN IN BAW-10040 (NO PUMP HEAD DEGRADATION AS $F(\alpha_F)$)	B&W AIR-WATER DATA (HEAD & TORQUE DEGRADATION)
PEAK SPEED	276% x N_R (C_D = 0.6)	253% x N_R (C_D = 0.6)
	---	405% x N_R (C_D = 1.0)

F-6

METHODS FOR PUMP OVERSPEED REDUCTION

- CREDIT FOR MECHANISTIC BREAKS
- USE PUMP TO MOTOR DECOUPLER
- USE BEST-ESTIMATE BREAKFLOW MODEL
- ELECTRICAL BREAKING
- MFG FLYWHEEL WITH STRONGER MATERIALS
- INCREASE/IMPROVE SURVEILLANCE OF FLYWHEELS