February 16, 1983

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island Nuclear Station, Unit No. 1)	)	
	)	

LICENSEE'S TESTIMONY OF ROBERT C. JONES, JR. IN RESPONSE TO ALAB-708 ISSUE NOS. 4 THROUGH 7 (ECCS EVALUATIONS AND BOILER-CONDENSER COOLING)

SUMMARY

This testimony responds to the Appeal Board's stated concerns with the B&W ECCS evaluations of small-break loss-of-coolant accidents and the efficacy of boiler-condenser cooling to remove decay heat at TMI-1 for those breaks for which it is predicted to occur.

The pre-TMI-2 accident analyses to demonstrate TMI-1 compliance with 10 C.F.R. § 50.46 used the NRC approved Appendix K model and, for certain break sizes, the results of these analyses also exhibited the steam generator heat transfer characteristics associated with boiler-condenser cooling.

The post-TMI-2 accident analyses used the approved CRAFT2 computer code, but modifications were made to the model to provide a more detailed examination of plant response under boiler-condenser conditions.

A revised B&W evaluation model, submitted to the Staff for Appendix K approval, has been used to analyze a 0.01 ft^2 break, during which boiler-condenser cooling is predicted to occur, and an extrapolation of the results demonstrates that adequate core cooling is maintained. While breaks smaller than the original spectrum (<u>i.e.</u>, 0.04 ft^2) do not need to be analyzed to demonstrate compliance with section 50.46, the response to NUREG-0737 Items II.K.3.30 and II.K.3.31 will provide further confirmation that the original spectrum analyzed was adequate (<u>i.e.</u>, that 0.07 ft^2 is the worst case).

The foregoing analyses demonstrate the adequacy of the boiler-condenser cooling mode to remove decay heat at TMI-1. A heat transfer analysis of the steam generator provides yet a further illustration of that capability. In addition, experimental data is discussed which supports this conclusion from the analyses.

INTRODUCTION

This testimony, by Robert C. Jones, Jr., Supervisory Engineer, Operational Analysis Unit, Babcock & Wilcox Company, is in response to Issue Nos. 4 through 7 of the Appeal Board's Memorandum and Order of December 29, 1982 (ALAB-708). Collectively, those issues address the adequacy of the B&W Emergency Core Cooling System (ECCS) evaluations of small-break loss-of-coolant accidents (small-break LOCAs) and the efficacy of boiler-condenser cooling to remove decay heat at TMI-1 for those breaks for which it is predicted to occur.

Licensee evidence in the record which is relevant to these issues, and which may provide valuable background information, includes:

- Licensee's Testimony of Robert W. Keaten and Robert C. Jones in Response to UCS Contention Nos. 1 and 2 (Natural and Forced Circulation), ff. Tr. 4588;
- Licensee's Testimony of Robert C. Jones, Jr. and T. Gary Broughton in Response to UCS Contention No. 8 and ECNP Contention No. 1(e) (Additional LOCA Analysis), ff. Tr. 5038;
- Licensee's Testimony of Robert C. Jones, Jr. and T. Gary Broughton in Response to the Board Question on UCS Contention 8, ff. Tr. 5039;
- Licensee Exhibits 3 through 13 (small-break LOCA and other accident analyses performed before and after the TMI-2 accident; small break operator guidelines).

ISSUE NO. 4: Whether the modified B&W ECCS evaluation model for small breaks that predicts the boiler-condenser process is an NRC approved code under Appendix K to 10 CFR Part 50 (from the staff).

RESPONSE

BY WITNESS JONES:

NRC regulations provide the definition of an ECCS evaluation model.

> An evaluation model is the calculational framework for evaluating the behavior of the reactor system during a postulated loss-of-coolant accident (LOCA). It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

10 C.F.R. § 50.46(c)(2).

Analyses performed prior to the TMI-2 accident to demonstrate the conformance of TMI-1 to 10 C.F.R. § 50.46 used the NRC-approved B&W ECCS evaluation model and, for certain break sizes (e.g., the 0.04 ft^2 break), the results of these analyses also exhibited the steam generator heat transfer characteristics associated with boiler-condenser cooling.

The model used for the additional small-break LOCA analyses performed after the TMI-2 accident that predict the boiler-condenser process technically was not the B&W ECCS evaluation model approved by the NRC pursuant to Appendix K to 10 C.F.R. Part 50. The model used for those analyses was the approved B&W evaluation model modified only by the addition of two control volumes (or nodes) to provide a more detailed examination of plant response under boiler-condenser conditions. No changes were made, however, to the CRAFT2 computer code, which is the approved Appendix K code used to predict system response for these breaks.

The additional control volumes, one in each Reactor Coolant System loop, were included in order to explicitly represent the upper head, or plenum, region of each steam generator. The analytical impact of the addition of the control volumes was to allow for a more accurate representation of the formation of a steam bubble between the steam generator emergency feedwater injection point and the 180° U-bend in the top of each RCS hot leg. See Licensee Ex. 5, § 6.2.4.2.

It should also be noted, as I discuss more fully below in response to Issue No. 7, that the B&W ECCS evaluation model for small-break LOCAs has been further revised, in response to Item II.K.3.30 of NUREG-0737. The changes made to the model include the addition of a steam generator upper head region, as discussed above, and others developed in consonance with the NRC Staff. The revised model has been formally submitted to the NRC (see Licensee Ex.) for review by the Staff for compliance with Appendix K to 10 C.F.R. Part 50.

ISSUE NO. 5: Whether the staff has reviewed the B&W Appendix K model to determine the ability of the code to calculate the effects of small breaks, including reliance upon boiler-condenser circulation (from the staff).

RESPONSE

BY WITNESS JONES:

While I obviously cannot describe the scope of the Staff's review beyond what the Staff itself has reported, as I indicated above the results of the analyses performed prior to the TMI-2 accident to demonstrate the conformance of TMI-1 to 10 C.F.R. § 50.46, with the approved B&W Appendix K model, exhibited, for certain break sizes, the steam generator heat transfer characteristics associated with boiler-condenser cooling.

The documentation of a revised B&W ECCS evaluation model, submitted to the Staff in November, 1982 under NUREG-0737 Item II.K.3.30 for review against Appendix K, includes the results of an analysis of the 0.01 ft^2 break, during which boiler-condenser cooling is predicted to occur. See Licensee Ex. ____ at Appendix E.

ISSUE NO. 6: Whether only breaks slightly smaller than 0.07 ft^2 must be analyzed (from the staff).

RESPONSE

BY WITNESS JONES:

The smallest break analyzed in the demonstration, prior to the TMI-2 accident, of TMI-1 conformance to 10 C.F.R. § 50.46 was of the size 0.04 ft^2. See Jones and Broughton, ff. Tr. 5038, at 12 (Table 1); Licensee Exs. 3 and 4. Breaks smaller than 0.04 ft^2 do not need to be analyzed to demonstrate the conformance of TMI-1 to section 50.46.

Section 50.46 establishes the criteria for an acceptable emergency core cooling system. Appendix K to 10 C.F.R. Part 50 sets forth the required and acceptable features of an evaluation model used to show compliance with 10 C.F.R. § 50.46. ECCS cooling performance is to ". . .be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated loss-of-coolant accidents is covered." See 10 C.F.R. § 50.46(a)(1).

B&W's selection of the spectrum of small breaks to be evaluated pursuant to 10 C.F.R. § 50.46 was based on the following considerations:

1. A Core Flood Tank (CFT) line break, by its location, severely limits the Emergency Core Cooling Systems

available for accident mitigation. Considerations of break location and single active failure dictate that core cooling must be provided by one high pressure injection (HPI) train and one core flood tank, until the active low pressure injection (LPI) train can be switched from its assumed injection into the broken CFT line and balanced between the two CFT lines. This break is analyzed, then, because it would appear to represent a limiting condition.

2. A series of break sizes are evaluated wherein the consequences of the rupture are mitigated by various combinations of the three ECCS systems:

   A. A break is considered for which mitigation is provided by the LPI, CFT and HPI systems.

   B. A break is considered for which mitigation is supplied by only the CFT and the HPI systems.

   C. A break is considered for which mitigation is provided solely by the HPI system.

Breaks are uniformly located, with the exception of the Core Flood line break, between the high pressure injection point in the cold leg (reactor coolant pump discharge piping) and the inlet to the reactor vessel. This location minimizes the amount of high pressure injection available for core cooling since a significant portion of the HPI flow can be discharged directly out the break. In addition, breaks at low elevations within the Reactor

Coolant System drain the Reactor Coolant System of significantly more water than breaks at higher elevations. Thus, for accidents in which the HPI or other ECCS systems cannot instantaneously provide core cooling and cooling must be sustained for some period of time via the initial RCS inventory, that inventory is reduced in the most rapid way possible.

3. Additional breaks are considered to confirm that the above spectrum has indeed bounded the worst case. That is, as necessary, break sizes smaller and larger than the calculated worst case are considered in order to confirm that the most adverse core cooling situation has been identified.

Very small breaks, i.e., those smaller than the smallest break considered in the spectrum (0.04 ft^2), are not evaluated because they are bounded by larger breaks for the following reasons:

1. Because of the internal vent valves, condensation within the steam generator must occur prior to uncovering of the reactor core. At TMI-1, this occurs because the injection location for emergency feedwater is near the top of the steam generator. Ultimately, the steam generator is filled to 95 percent on the operating range, which assures a condensing surface above the top of the core continuously.

2. If steam condensation is occurring in the primary side of the steam generator, then the RCS pressure will be reduced to near the pressure of the secondary side of the steam generators (approximately 1000 psi) or at a higher pressure wherein the HPI flow matches the leak flow.

3. The breaks evaluated in the spectrum, those with HPI mitigation only, drain the RCS loops faster and establish steam condensation earlier than do smaller breaks. At the start of the steam condensation mode, the decay heat rate for the larger break will be higher than for the smaller break. The larger break will also be losing initial RCS inventory faster than the smaller break. Thus the potential for core uncovery is greater for the larger breaks.

4. Because it has been shown by evaluation that the HPI provides successful mitigation of a transient at a higher decay heat rate at an earlier time, the HPI will provide successful mitigation of the transient at a lower, later decay heat rate. Therefore, smaller breaks cannot have consequences in the core region more severe than the smallest break considered in the spectrum evaluation.

Therefore, while breaks smaller than the spectrum analyzed to demonstrate compliance with 10 C.F.R. § 50.46 may involve different system behavior (i.e., the repressurization cycle which is caused by the interruption of natural circulation), core cooling is dependent upon maintaining core coolant inventory. Regardless of the specific sequence of events during a very-small-break LOCA, before core uncovery can occur, reactor coolant pressure will decrease to a point (approximately 1000 psig) where high pressure injection has been demonstrated to provide adequate core cooling for the maximum core decay heat level.

The additional small-break LOCA analyses performed after the TMI-2 accident provided further confirmation of the validity of the above described methodology. While these evaluations were for the purpose of providing an improved analytical basis for emergency operating procedures, rather than to demonstrate compliance with 10 C.F.R. § 50.46, several breaks smaller than the previously analyzed 0.04 ft^2 break were addressed. Specifically, breaks of 0.005 ft^2 and 0.01 ft^2 were evaluated. See Jones and Broughton, ff. Tr. 5038, at 6-7 and 17 (Table 6). In my opinion, the analyses for the 0.005 ft^2 and 0.01 ft^2 breaks are sufficient to demonstrate conformance to 10 C.F.R. § 50.46 pursuant to Appendix K. The results indeed showed that, compared to the larger break sizes, an increased margin relative to core uncovery existed. The effort now underway, pursuant to NUREG-0737 Items II.K.3.30 and II.K.3.31, to analyze small breaks with an improved Appendix K model, is aimed at providing yet further confirmation that the original spectrum of breaks analyzed was adequate to demonstrate conformance to 10 C.F.R. § 50.46.

ISSUE NO. 7: Confirmation (such as by means of detailed computational analysis or experimental testing) that boiler-condenser circulation flow will transport sufficient core decay heat to the steam generators to prevent core damage (from the licensee and the staff).

RESPONSE

BY WITNESS JONES:

For certain sized small-break LOCAs, the steam generators are necessary to remove a portion of the decay heat added to the primary system.1/ The Appeal Board has questioned the adequacy of energy removal via the steam generators while operating in the boiler-condenser mode of cooling. Additional analyses are presented in this testimony to demonstrate that boiler-condenser heat removal at TMI-1 is sufficient to remove core decay heat following a LOCA. I have also provided a discussion of the experimental data which supports this conclusion from the analyses.

Before discussing the boiler-condenser mode of cooling, however, it is necessary to discuss the relationship between energy removal from the fuel rods (core cooling) and energy removal from the reactor coolant system (RCS). To ensure adequate core cooling during a small-break LOCA, it is

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1/ The discussion that follows assumes the availability of emergency feedwater and one HPI train. Steam generator heat removal is not necessary if two HPI pumps are available. See Jones and Broughton, ff. Tr. 5038.

necessary to maintain a two-phase level within the reactor vessel which is at or near the top of the core. In this manner, the core decay heat which is being generated can be removed from the fuel rods by pool boiling or, if the core is slightly uncovered, by forced convection to superheated steam. The HPI system has been designed to provide the necessary fluid makeup to the RCS to ensure adequate core heat removal.

Decay heat removal from the RCS can be accomplished in several ways, e.g., by break flow, steam generator heat removal, or combinations thereof. During a small-break LOCA, the decay heat removal is important in that it determines the system pressure and, hence, the HPI flow being provided. Therefore, to demonstrate core cooling, it is only necessary to show that sufficient decay heat removal is provided, prior to core uncovery, to allow the HPI system to replace the inventory being boiled by core decay heat removal. In this manner, level in the core can be maintained above the top of active fuel rods.

For break sizes smaller than 0.02 ft^2, decay heat removal from the RCS is accomplished by a combination of the break flow and the steam generators. See Keaten and Jones, ff. Tr. 4588, at 7. If the break sizes are smaller than 0.005 ft^2, the HPI system can compensate for the break flow and maintain the primary coolant loops essentially full of liquid such that natural circulation is not interrupted.

Assuming a break size between 0.005 and 0.02 ft^2, the HPI flow is unable to compensate for the leak flow and the RCS will saturate. Steam pockets will eventually form and grow to a volume sufficient to fill the 180° inverted U-bends at the top of both hot legs. This will result in an interruption of natural circulation. The loss of natural circulation leads to a loss of heat removal via the steam generators and the system will pressurize. See Jones and Broughton, ff. Tr. 5038, at 6-7; Keaten and Jones, ff. Tr. 4588, at 7.

As the RCS continues to lose inventory, a condensing surface will be exposed in the steam generators. This will establish the boiler-condenser mode of heat removal. This mode of heat removal will terminate the pressure increase and control RCS pressure at a value sufficient to assure adequate HPI flow for core cooling. See Jones and Broughton, ff. Tr. 5038, at 6-7.

Small-break LOCA analyses have been performed which demonstrate the adequacy of this cooling mode. These are documented in Licensee's Exhibit 5. Those analyses were performed utilizing the presently approved CRAFT2 code. Comparison of the steam generator heat removal rates calculated in those analyses to that which would be obtained by using the theoretical formulations in the new model show reasonable agreement. That is, an approximate three-foot adjustment in the condensing length would yield the same heat transfer. This small loss of inventory, approximately ten percent of the available inventory above the top of the core, would not affect core cooling.

Since the analyses in Licensee's Exhibit 5, the B&W ECCS evaluation model and the CRAFT2 code have undergone modification in response to II.K.3.30 of NUREG-0737. The revised evaluation model and CRAFT2 code have been submitted to the NRC for review.

Within the modified CRAFT2 code, an upgraded steam generator model has been incorporated which includes heat transfer correlations specifically oriented to the boiler-condenser mode of cooling. A new 0.01 ft^2 break analysis has been performed using the revised code and is documented in BAW-10154. See Licensee Ex. ____, Appendix E. Extrapolation of the results demonstrate that adequate core cooling is maintained for breaks of the size for which boiler-condenser cooling is predicted to occur.

The capability of the steam generator to remove sufficient core decay heat to assure adequate core cooling via the HPI system during a small break LOCA is further illustrated by the analysis described below. As stated previously, adequate core cooling is assured if the core is continuously covered by a two-phase mixture. Maintenance of this condition is assured if the HPI flow provided to the system is sufficient to match or exceed the inventory boiled off from core decay heat removal.

Because the HPI flow varies with system pressure, the time at which the injected flow and core boiling match will be a function of the system pressure. The pressure/time relationship for this matchup is illustrated on Figure 1. Thus, the significant question is whether the boiler-condenser mode will assure a pressure/time relationship, before the core becomes uncovered, to yield adequate HPI to keep the core covered.

A heat transfer analysis of the steam generator, while operating in the boiler-condenser mode, was performed to develop the pressure/time relationship. Prior to any possible uncovering of the core, the full condensing surface of the steam generator will be exposed. Using this surface area, an analysis was performed to determine the RCS temperature, and hence pressure, necessary to condense all the steam being generated as a result of core decay heat removal as a function of time. It should be noted that since none of the generated steam is assumed to be removed via the break, this analysis would overpredict the RCS pressure that could exist just prior to possible core uncovery. Figures 2 and 3 show the results of the steam generator heat removal analysis for cooling on the steam generator level (at 95 percent on the operating range) and the emergency feedwater spray, respectively.

Combining the results of the HPI cooling and steam generator heat removal analyses, as illustrated in Figure 4, it is seen that boiler-condenser heat removal will provide sufficient pressure control to result in HPI flows necessary to assure adequate core cooling after 1650 seconds. The next subject of the analysis, then, is to determine whether the core is predicted to become uncovered prior to this time.

Several small break LOCA analyses have been performed which indicate that the core could not become uncovered prior to 1650 seconds for the break sizes of interest. In Licensee's Exhibits 3 and 4, which are the section 50.46/Appendix K analyses for TMI-1, it can be seen that the 0.04 ft^2 break reaches its minimum system inventory at 3000 seconds. No uncovering of the core is calculated for this break. Since smaller breaks would lose inventory at a slower rate, the 0.04 ft^2 break would bound the results.

In addition, the analyses of the 0.01 ft^2 break (documented in Licensee's Exhibit ___ (BAW-10154), show that the boiler-condenser mode of cooling is calculated to occur at approximately 1500 seconds. At this time, there is a substantial quantity of liquid (105,600 lb or 2440 ft3) remaining above the top of the core. This inventory would have to be lost through the break prior to the core uncovering. Even if an RCS pressure of 2500 psi was assumed, which is well above the 1800 psi pressure calculated for this time, this inventory could not be lost prior to 1650 seconds.

Based on this analysis, it is clear that uncovering of the core would not occur prior to 1650 seconds for the break size range for which boiler-condenser heat removal is necessary. Since the boiler-condenser cooling mode assures adequate pressure control after this time to enable the HPI to match or exceed the core boil-off, adequate core cooling is assured.

Turning to the Appeal Board's interest in experimental testing of the boiler-condenser mode of heat removal, it should be recognized that the actual heat transfer mechanisms are well understood. Within the primary system steam is condensed on the inside wall of the cooled steam generator The heat then flows through the tubes, via conduction, and is transferred to the secondary side fluid. Two possible mechanisms exist for the secondary side heat transfer. These are by pool boiling on the immersed steam generator tubes and/or cooling by the emergency feedwater which is sprayed directly on the steam generator tubes.

There are several data sources available, or planned, which demonstrate the capability of the steam generator to remove heat in a boiler-condenser mode. First, there is the TMI-2 accident itself. After all of the reactor coolant pumps had been tripped at 100 minutes, filling of the steam generator by emergency feedwater commenced. During the fill period, heat removal from the RCS occurred which controlled the primary system pressure within 100 psi of the secondary side pressure. The only explanation for the pressure curves tracking together is the effect of boiler-condenser cooling in removing decay heat. See UCS Ex. 1 (minutes 100 to 125). If the HPI system had been actuated and maintained at this time, adequate inventory would have been maintained to prevent core damage. Thus, the TMI-2 accident did not demonstrate an inadequacy of RCS heat removal (i.e., an inadequacy of boiler-condenser cooling), but rather showed the importance of maintaining adequate core inventory via the HPI.

Tests have also been run at the Alliance Research Center (ARC) which examined condensation phenomena in a high pressure facility. In these tests, a single steam generator tube was tested by exposing a condensing surface by adjusting water level on the inside surface of the tube. Then, by varying steam flow to the test section, temperature measurements were taken in order to determine the heat transfer coefficient. The calculated coefficients for these tests have confirmed the conservatism of the heat transfer model employed in the upgraded CRAFT2 code.

In the future, additional experimental data on the boiler-condenser mode of cooling and small break LOCA response will be developed at ARC. At present, an integrated systems test facility at ARC (GERDA) is being tested. It is a scaled single-loop, full height, full pressure test facility of a B&W NSS and is of similar size to Semiscale. This facility was developed for the BBR company in Germany in order to examine small break LOCA phenomena. The data from this facility is expected to be available in mid-1983.

The B&W Owner's Group, in conjunction with the NRC, is presently exploring a two-loop facility to further examine plant response to small break LOCA and other transients. This data will be used to confirm the adequacy of the computer models. Through the computer codes, this data will then enhance the understanding of plant response for improved operator training and procedures. Data from this facility is projected to be available in mid-1985.

In summary, the boiler-condenser mode of cooling is relied upon for heat removal during certain sized small break LOCAs. The basic heat removal processes are well understood and have been successfully applied in other engineering applications. The ability of the TMI-1 steam generator to remove core decay heat has been demonstrated as sufficient to provide adequate core cooling. Thus, while there are presently plans to obtain additional experimental data for the purposes of improved understanding of plant response and for code benchmarking, operation of TMI-1 prior to receipt of this data will not endanger the public health and safety.

Figure 1

HPI MATCHUP WITH CORE DECAY HEAT

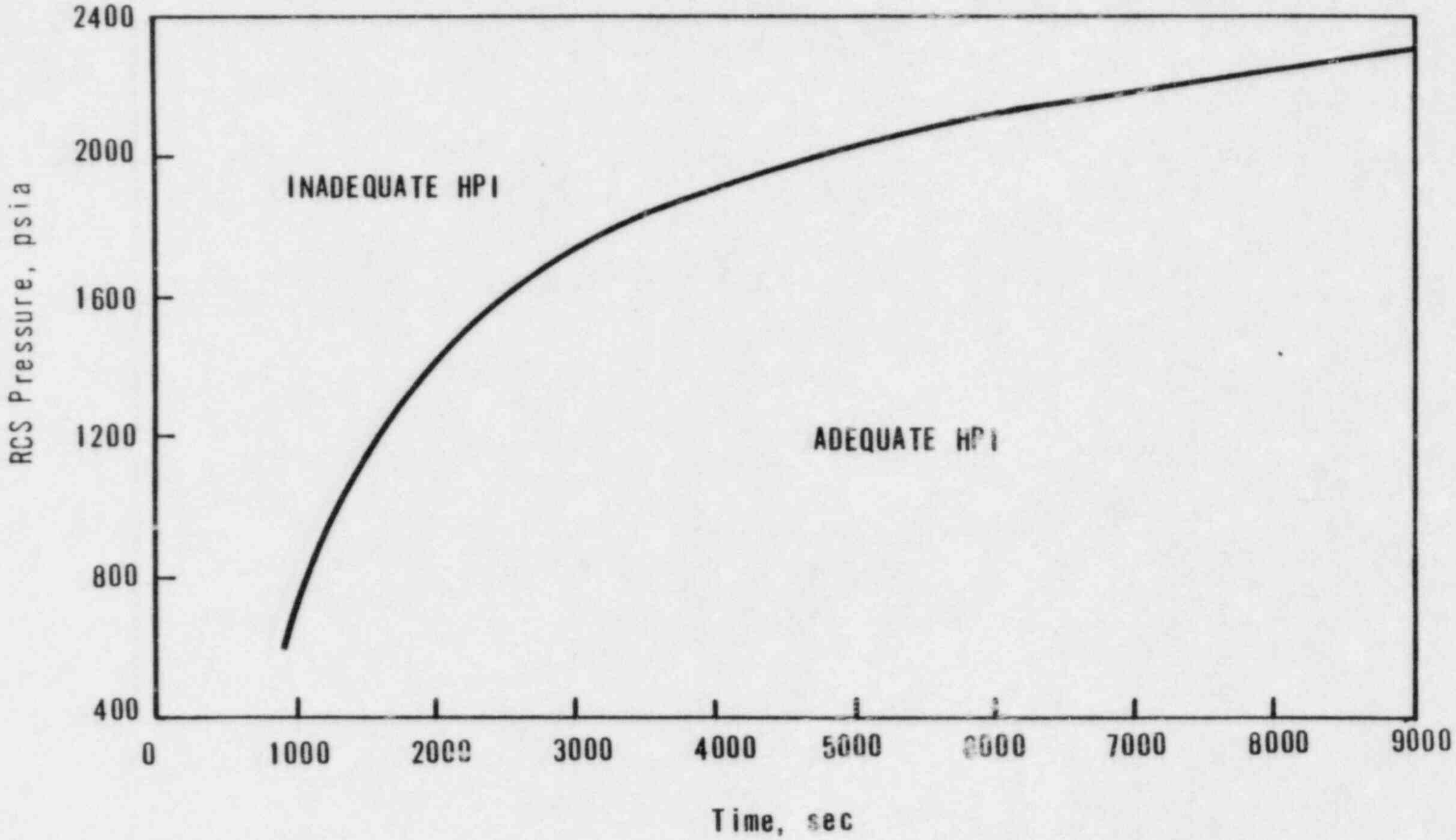


Figure 2

RC PRESSURE VS TIME FOR BOILER-CONDENSER HEAT REMOVAL VIA SG LEVEL

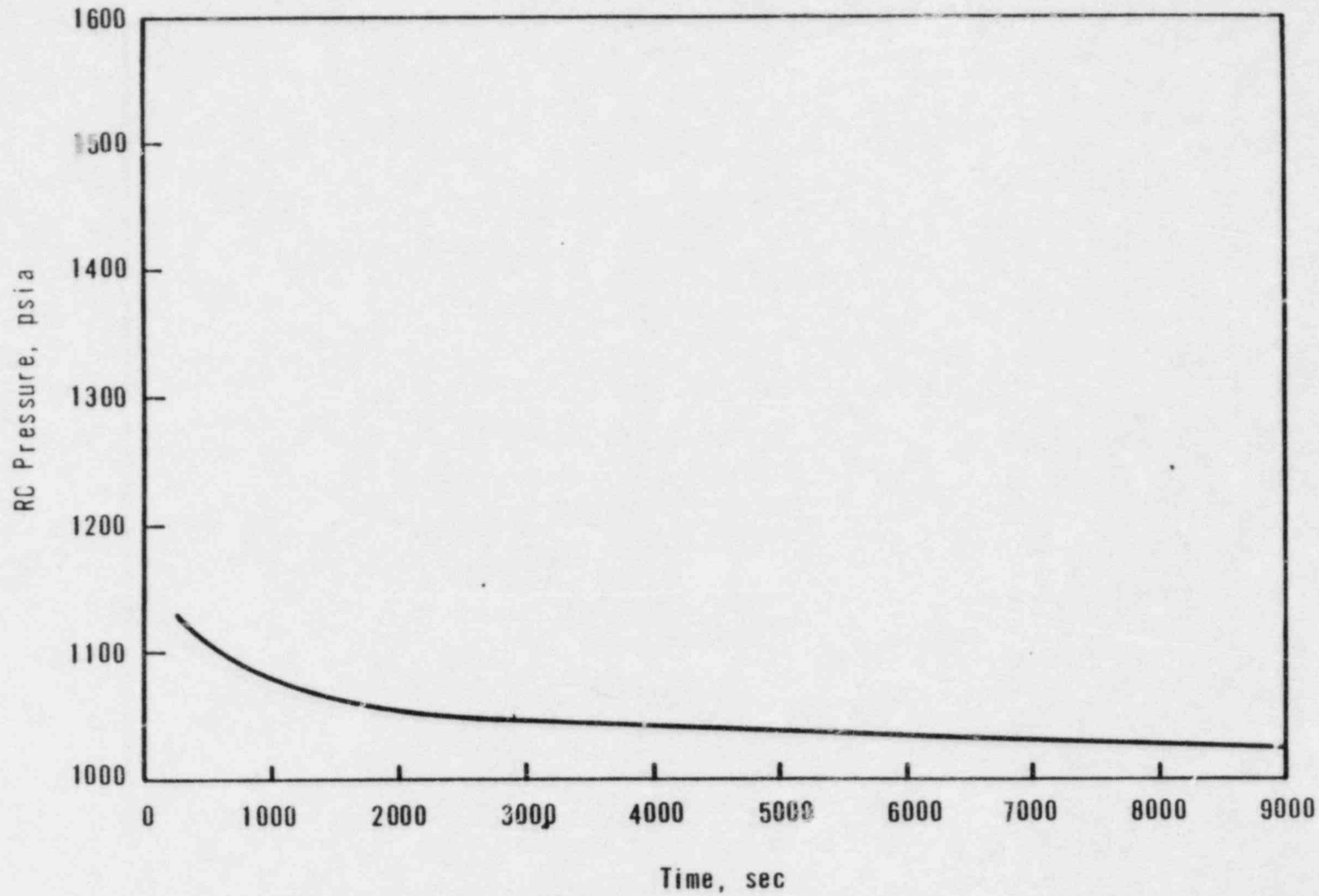


Figure 3

RC PRESSURE VS TIME FOR BOILER-CONDENSER

HEAT REMOVAL VIA EFW SPRAY

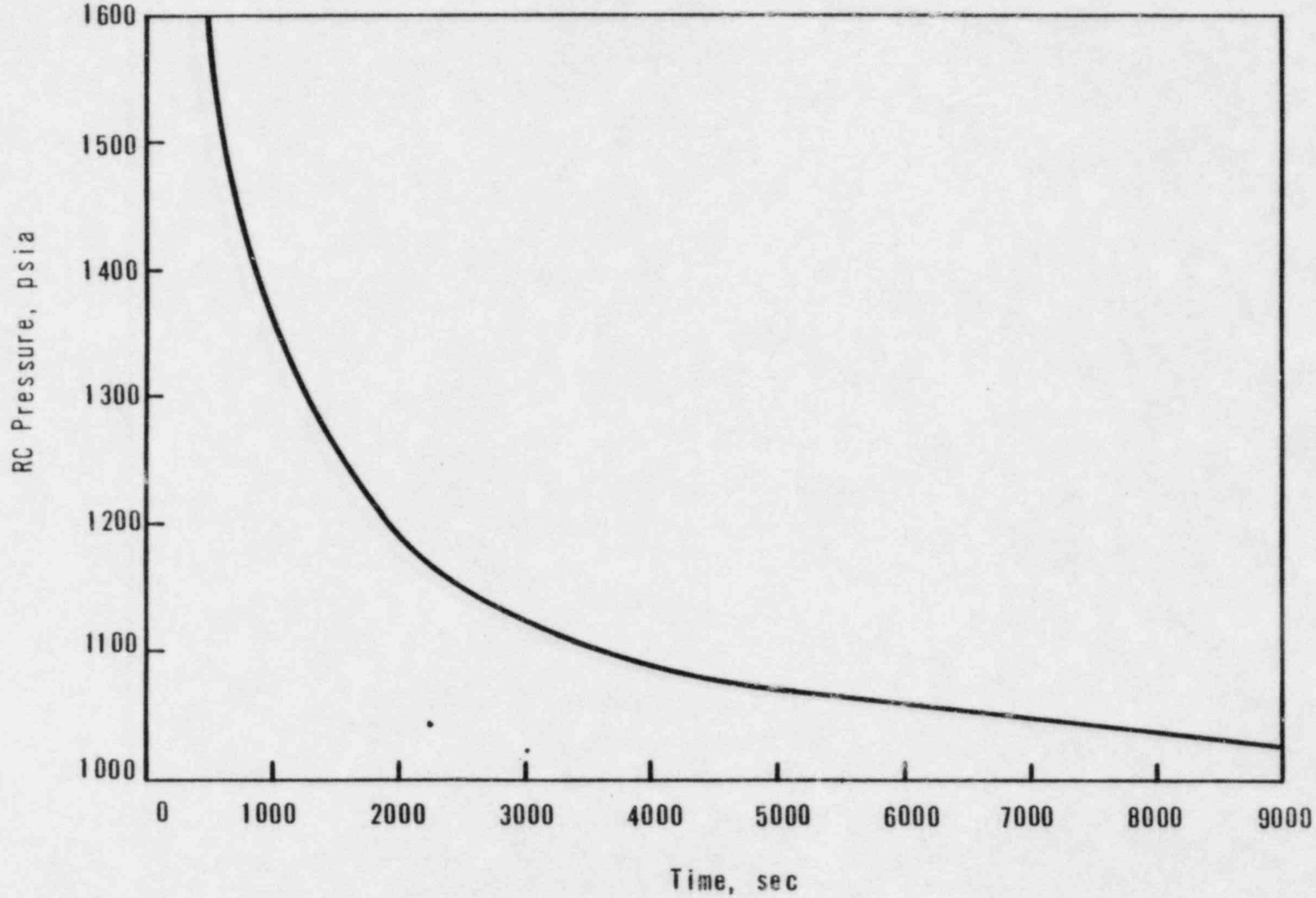
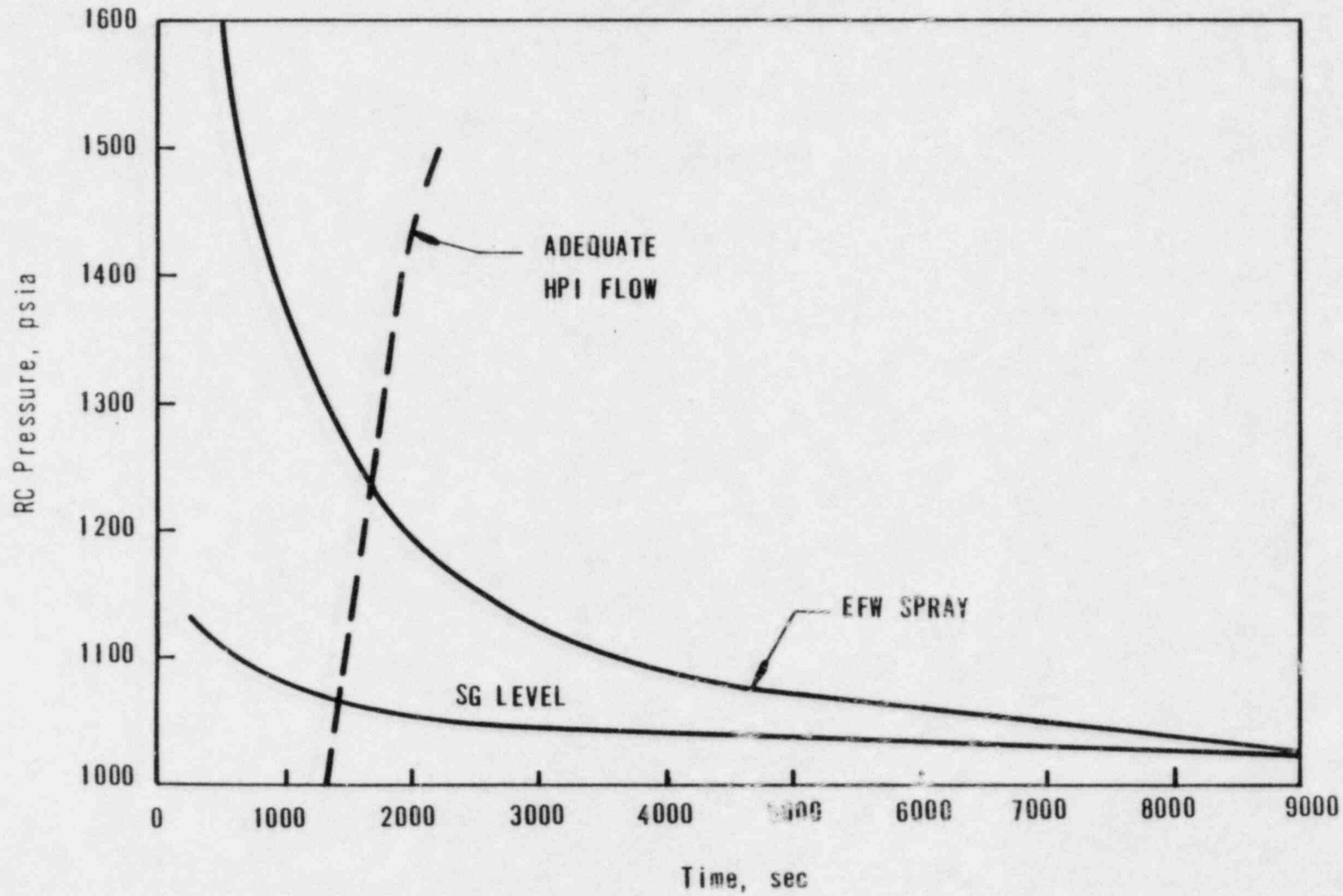


Figure 4

RELATIONSHIP OF HPI COOLING AND BOILER CONDENSER HEAT REMOVAL



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Education: B.S., Nuclear Engineering, Pennsylvania State University, 1971. Post Graduate Courses in Physics, Lynchburg College.

Experience: July 1982 to present: Supervisory Engineer, Operational Analysis Unit, B&W. Responsible for the performance of plant transient analyses and analyses used in the development of operator guidelines. During this period, has continued as Project Engineer for B&W analyses performed in response to NUREG-0737 Item II.K.3.30.

June 1975 to July 1982: Acting Supervisory Engineer and Supervisory Engineer, ECCS Analysis Unit, B&W. Responsible for calculation of large and small break ECCS evaluations, evaluations of mass and energy releases to the containment during a LOCA, and performance of best estimate pretest predictions of LOCA experiments as part of the NRC Standard Problem Program. Involved in the preparation of operator guidelines for small-break LOCA's and inadequate core cooling mitigation.

June 1971 to June 1975: Engineer, ECCS Analysis Unit, B&W. Performed both large and small break ECCS analyses under both the Interim Acceptance Criteria and the present Acceptance Criteria of 10 CFR 50.46 and Appendix K.