Enclosure

# SAFETY EVALUATION REPORT
# MODIFICATIONS TO THE BOILING WATER REACTOR (BWR) EMERGENCY PROCEDURE GUIDELINES TO ADDRESS REACTOR CORE INSTABILITIES

## 1.0 SUMMARY

This safety evaluation report (SER) documents the staff's review of the modifications proposed for the Emergency Procedure Guidelines (EPGs), Revision 4 by the Boiling Water Reactor Owners' Group (BWROG). These revised EPGs implement the changes reviewed and approved in the staff's February 1994 SER[1] to mitigate the consequences of power instabilities during anticipated transients without scram (ATWS). NRC focused its review on two additional aspects of the EPGs: (1) the optimal water level control strategy and (2) the effectiveness of boron remixing. The staff concluded that rapid boron remixing (less than 10 minutes) will likely occur with flow rates above 15 percent (not 5-10% as previously assumed) of rated flow. The staff also concluded that licensees should consider strengthening the procedural guidance by specific instructions to ensure early boron injection after confirmation of ATWS and initiation of actions to lower water level, and by a higher level control strategy (targeting TAF[*]+5 feet (1.52m), but not to exceed 2 feet (0.61m) below the feedwater sparger) for plants that inject boron through a standpipe below the core.

The staff concludes that the suggested change will reduce the uncertainties regarding achieving reactor shutdown without depressurization and will, overall, reduce core damage risk. However, the staff also concludes that the mixing of stratified boron using the lower level strategy of the BWROG guidance will most likely be adequate to avoid reactor depressurization. The staff bases this expectation on the results of scale-model experiments, plant transient data, and calculations and natural circulation data indicating that a high core flow can be expected to promote destratification of cold boron solution when using the BWROG procedural guidance. The staff has also considered the risk aspects for

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[*]Top of active fuel

this event and the incremental risk improvement that might be achieved by requiring a change to the BWROG level control guidance. The staff finds that risk considerations do not justify imposing a backfit to incorporate the higher water level control strategy preferred by the staff review. Cost considerations are the BWROG burden for changes in training and procedures and the staff effort to oversee the implementation. It is anticipated that the staff resources needed to ensure a timely involuntary implementation of new procedures would be substantial in comparison to the minimal audit effort needed to monitor voluntary conformance by BWROG licensees to the EPGs. Therefore, the staff has modified its draft SER to avoid imposing its findings as conditions of its approval. The BWROG guidance is acceptable, but deviations from the guidance that are consistent with the staff's recommendations in the SER are also acceptable and encouraged.

## 2.0 INTRODUCTION

After the 1988 instability event at LaSalle Unit 2, the staff of the U.S. Nuclear Regulatory Commission (NRC) (aided by its contractors - - Oak Ridge National Laboratory (ORNL), Brookhaven National Laboratory, Idaho National Engineering Laboratory), and BWROG (aided by its contractors General Electric Co. and Operations Engineering, Inc., (OEI)) began a concentrated effort to improve the understanding of and response to boiling-water reactor (BWR) thermal-hydraulic instability with an ATWS event. These studies showed that instabilities were mitigated by rapidly lowering the water level in the reactor vessel below the water level in the feedwater sparger. This action reduces the core inlet subcooling by preheating the coolant with condensing steam in the steam dome. The staff approved this approach in its SER of February 5, 1994,[1] by approving NEDO-32047[2] and NEDO-32164.[3]

To improve operator guidance for an ATWS event, BWROG changed Revision 4[4] of the Emergency Procedures Guidelines to be consistent with the instability mitigation strategy. This effort included analytical studies to determine the overall strategy for controlling water level so as to effectively shut down the reactor during an ATWS by injecting of soluble poison. In a letter dated March 21, 1994,[5] BWROG requested approval of its proposed EPG modifications. In a letter dated April 25, 1994, BWROG submitted a report prepared by OEI[6] that contained the analytical support for the ATWS mitigation strategy proposed by BWROG. The report was revised and resubmitted in September 1994 after its validation was completed.[7]

In an August 2, 1994, meeting between the NRC staff, the BWROG EPG committee, and the Pennsylvania Power and Light Company (PP&L), licensee for the Susquehanna BWR plants, BWROG recommended that the staff approve a deviation from the EPGs in the water level control strategy used by PP&L.[8] PP&L prefers a higher level control strategy (Strategy A) to reduce the risk of core uncovery and fuel damage associated with lower water level control. BWROG recommended that PP&L be permitted its proposed plant-specific deviation because both strategies are acceptable and there is insignificant difference in risk.

By letter dated June 16, 1995, the staff transmitted to R. A. Pinelli, Chairman of the BWROG, its draft safety evaluation report[9] (DSER) accepting the BWROG proposed EPG modifications. The letter requested comments on the DSER because the staff had conditioned its acceptance on some changes to the guidance proposed by BWROG. The changes, based on the staff review including audit calculations, involved earlier boron injection and target water level control at a higher level than that proposed by the BWROG. The DSER concluded that the revised guidance would reduce the heat load of the suppression pool during a worst case ATWS event, thus reducing the dependence on BWROG boron mixing assumptions to ensure that emergency depressurization of the reactor to protect containment integrity would not be required by procedure. It is desirable to avoid reactor depressurization during an ATWS event because it results in increased core damage risk. The DSER was also placed in the Federal Register and comments were invited from other interested parties.

At an August 9, 1995, NRC/BWROG management meeting to discuss the DSER, it was agreed that the differences in the appropriate level control strategy in the EPGs were dictated by the effectiveness of remixing of cold boron solution injected into the reactor coolant through a perforated standpipe in the reactor vessel below the core region. The EPGs require the reactor water level to be lowered in order to reduce core flow and reactor power while enough boron solution to shut down the reactor is being injected. At a low core flow threshold, the boron solution ceases to mix with the circulating coolant and stratifies in the lower reactor vessel head. After a predetermined quantity of boron has been injected, the water level in the reactor is raised to the normal operation level, thus increasing core flow and promoting remixing of the stratified boron with core coolant to shut down the reactor. The EPGs are designed to accomplish the reactor shutdown before the pressure suppression pool is heated to its temperature limit by steam exhausted through the reactor safety relief valves. On September 15, 1995, BWROG submitted its comments on the DSER[10] and additional information describing the boron remixing tests and plant thermal stratification experience. The BWROG submittal is addressed in Appendix A.

In this safety evaluation, the NRC staff assesses the acceptability of the BWROG-proposed modifications to the EPGs for ATWS instability mitigation and reactor shutdown via water level control and boron injection. Staff review includes consideration of the information provided in the response to the DSER. The acceptability of the plant-specific deviation requested for the Susquehanna units is also considered.

## 3.0 EVALUATION

The NRC contractor, ORNL, assisted the staff in the overall evaluation by reviewing the BWROG submittals documenting the proposed EPG modifications and the analytical support. ORNL also assisted the staff in its audit of the OEI model, Containment Response Analysis Code (CRAC), which was used to perform the BWROG calculations. ORNL prepared a technical evaluation report (TER), which is attached, documenting its findings regarding the effectiveness of the proposed EPG modifications and the results of the audit of CRAC, including audit calculations performed by the staff. Staff calculations were performed using TRAC-BF1[11] (with 1-D neutronics) and RAMONA-4B[12] to evaluate the relationship between water level, power, and core flow during water level reduction and boron injection through a standpipe below the core. Results obtained with these two codes were reasonably consistent and were also comparable to PP&L findings for Susquehanna BWRs using SABRE, after making adjustments to compensate for different procedural assumptions. All of the analyses, except the BWROG CRAC calculations, produced results that favored a high (5 feet above the top of active fuel) water level control strategy over the lower (below the top of active fuel) water level control strategy for BWRs designed to inject boron into the lower plenum. The staff audit of the CRAC code, which is discussed in Appendix A of the TER, concluded that CRAC predicts non-physical results. For BWRs designed to inject boron above the core where stratification at low core flow is not a factor, the lower level control strategy (Strategy B) results in reduced power with less cumulative heat load to the containment.

Additionally, the staff considered the plant-specific deviation in level control strategy requested for the Susquehanna plants which inject boron solution through a standpipe below the core. The staff could find no plant-specific justification for this deviation for the Susquehanna units from other standpipe boron injection plants. However, the staff evaluation of the two competing strategies for level control and boron mixing concluded that the BWROG strategy for deliberate reduction in water level below the top of active fuel (TAF) results in no significant reduction in the cumulative heat load challenge to containment integrity when the standby liquid control (SLC) system fails to function. When the SLC system functions, both TRAC-BF1 and RAMONA-4B staff calculations indicate a significant advantage for the high-level strategy (Strategy A) because boron continues to mix to near shutdown concentrations before core flow is reduced to the stagnation threshold (core flow < 5%). Most calculations show a substantial margin to the suppression pool temperature limit at the time the hot shutdown boron weight (HSBW) has been injected and the remixing of stratified boron commences.

The staff review discovered an uncertainty associated with the sensitivity to the time required to remix stratified boron sufficiently to shut down the reactor. Additionally, the staff evaluation of the two competing level control strategies indicates that the lower level requires faster boron remixing once HSBW is injected because the suppression pool temperature is higher at that time. Because of the higher pool temperature when remixing begins and the uncertainty about the remixing time constant, the procedural guidance associated with the lower level control strategy (Strategy B) is more likely to result in suppression pool temperatures requiring reactor depressurization for BWRs designed to inject boron through a standpipe below the core. The staff believes it is important to maintain a low probability of conditions that will require depressurization of the reactor coolant system. This is because neither BWROG nor the staff evaluation provide rigorous evidence that core coolability and containment integrity can be ensured for that circumstance.

Risk Considerations

In the DSER,[10] the staff proposed a requirement that the licensees implement the high-level strategy in the emergency operating procedures for ATWS response. This action is contrary to the previously approved Revision 4 EPGs and a requirement to implement the high-level strategy would be a backfit in accordance with 10 CFR 50.109. To evaluate proposed backfits, the staff uses NUREG/BR-0058, "Regulatory Analysis Guidelines of the NRC."[13] The guidance in NUREG/BR-0058 helps the staff determine whether proposed backfit actions are needed and justified. The guidelines also establish a framework for (1) identifying the problem and associated objectives, (2) identifying alternatives for meeting the objectives, (3) analyzing the consequences of alternatives, (4) selecting a preferred alternative, and (5) documenting the analysis in an organized and understandable format. The latest NUREG/BR-0058 revision (Revision 2) contains the following: (1) NRC's accumulated experience with implementing Revision 1 of the guidelines and (2) changes in NRC regulations and procedures since 1984, especially the backfit rule, 10 CFR 50.109, and the policy statement on safety goals for operating nuclear power plants.

The guidelines in NUREG/BR-0058 have safety goal screening criteria that are intended to eliminate some proposed requirements from further consideration because core damage frequency (CDF) reductions are small. In Section 3.3.1 of NUREG/BR-0058, the staff states that for proposed regulatory actions to prevent or reduce the likelihood of sequences that can lead to core damage events, the change in the estimated CDF per reactor-year must be evaluated and addressed. As illustrated in Figure 3.2 of the guidelines, regulatory initiatives involving new requirements to prevent core damage should result in a reduction of at least 1.E-6 in the estimated mean value CDF in order to justify proceeding with further analyses. Regulatory initiatives with reductions in CDF below 1.E-6 are, according to the guidance, not to be pursued as backfits.

Staff Actions Pursuant to NUREG/BR-0058

The staff approached the analysis of the effect of ATWS on CDF in two ways. First, the staff performed a qualitative risk analysis based upon the results of its deterministic analyses. The purpose of this analysis was to identify the best method of comparing the two strategies before a quantitative analysis was attempted. The staff determined that the best method of comparison was to estimate the different level control failure probabilities using the two different strategies and to use these probabilities to evaluate the CDF with and without SLC. The results of the staff's quantitative bounding study and the qualitative argument used in the development of the bounding study follow. The bounding analysis used the following assumptions:

1. The ATWS initiating event probability is 3.E-5[14].
2. Emergency depressurization leads to core damage.
3. The failure probabilities are as shown in Table 1.

The bounding analyses performed considered two different scenarios and calculated the difference in CDF using the two level control strategies to manage the two different sequences. First, it was assumed that SLC had failed and that the plant is responding as shown in the upper set of curves (curves 1 and 2) in Figure 1 in the attached TER. The benefit of MSCWL control under SLC failure conditions is, as shown in Figure 1 of the TER, that controlling the level below TAF during the event will allow additional recovery time on the order of a minute before emergency depressurization is required. It is the judgement of the staff that this difference is insignificant and that both level control strategies will, with the same probability, lead to emergency depressurization when SLC fails. The basis for this judgement is that no significant recovery actions could be taken in the additional time available using MSCWL control to otherwise alter the outcome of the event. This conclusion is supported by Figure 12-3 in NUREG-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications."

**Table 1** Assumed Failure Probabilities for Staff Bounding Analysis

| Event Tree Branch | Failure Probability[*] |
|---|---|
| Primary Coolant System[+] | 1.0 |
| Standby Liquid Control System[#] | 0.17 |
| Level control at Top of Active Fuel + 5 feet | 0.01[&] |
| Level control at Minimum Steam Cooling Water Level | 0.05[?] |

[*]The failure probabilities listed only account for the operator's ability to successfully control level; they are not core damage frequencies.

[+]Condenser failure

[#]Considers both equipment and personnel failure. This value is not explicitly used in the analysis. It is included to show that SLC failure sequences have CDF valued on the same order of magnitude as SLC operable sequences.

[&]This value is from the referenced event tree model. It is supported by values from NUREG-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications."

[?]This value was calculated by assigning the more difficult level control strategy, MSCWL control, a value equal to the upper uncertainty bound used in NUREG-1278. This means that MSCWL control is assumed to be 5 times harder than TAF + 5 feet control.

The staff then considered the more likely event in which SLC functions. This case is illustrated by the lower set of curves (curves 3 and 4) in Figure 1 of the attached TER. This study is based upon an event tree model from NUREG/CR-4674, "Precursors to Potential Severe Core Damage Accidents: A Status Report," Volume 17, Page A-5, 1992. For this scenario, the CDF using TAF+5 feet control is 3.E-7, and with MSCWL control it is 1.5E-6. The difference in CDF between the two strategies is 1.2E-6. The CDF is computed by multiplying the initiating event frequency by the relevant failure probabilities.

In the qualitative probabilistic study performed by the staff, it was concluded that operator actions for Strategy A are simpler than actions for Strategy B because the reactor water level need not be reduced to the minimum steam cooling water level and level control requires less operator attention, leading to a slight reduction in CDF for Strategy A. Furthermore, Strategy A mixes more boron before it begins to stratify in the lower plenum as a result of low flow, and the integrated reactor power for the duration of the event is reduced compared to Strategy B. Heat load to the suppression pool using Strategy A is reduced when compared to Strategy B, resulting in, once again, a slightly lower risk of RPV emergency depressurization. If RPV emergency depressurization is required, there is great uncertainty about adequate core cooling to avoid core damage during the period of core uncovery. Using this qualitative argument, when the strategies are compared, the CDF is slightly less for the high-level operation when there is greater certainty that RPV depressurization can be avoided and level control is simpler. However, the bottom-line conclusion is that both strategies lead to success with boron injection but that the higher level strategy allows more room for uncertainties, such as operator response time and boron remixing time. (For a detailed description of boron remixing, refer to page 9 of the attached TER.)

The staff considered both of these analyses in its review. The staff concluded that high-level (Strategy A) control will lead to a maximum decrease in CDF on the order of 1.2E-6. NUREG/BR-0058 specifies that for modifications in which the estimated reduction of CDF is less than 1E-6, no action is required. Because of the bounding nature of the staff's analysis, it was concluded that a modification of the water level control strategy does not meet the "substantial additional protection" requirement of 10 CFR 50.109 to support a backfit. However, the staff urges BWROG to adopt high level control because it is simpler for operators to use and it allows more time to account for uncertainties in boron remixing and operator response time.

## 4.0 CONCLUSIONS

The staff reviewed the proposed EPG change package to improve the ATWS response and the alternate ATWS management strategy requested for one licensee. On the basis of the technical conclusions presented in the attached TER and the staff's own calculations, the staff concludes the following:

(1) The following two steps added to the EPGs to implement the strategy previously approved[1] to prevent or mitigate the consequences of reactor instabilities are acceptable:

   (a) Water level is reduced to at least 2 feet (0.61 m) below the feedwater sparger.

   (b) Early soluble boron injection is permitted on confirmation of an ATWS and required when large power oscillations are observed.

(2) As discussed in the TER, the reduction in hot shutdown boron weight from approximately 755 ppm hot to approximately 478 ppm hot (355 ppm referenced to the density of light water at cold shutdown conditions) to reduce the necessary injection time is acceptable.

(3) Allowing reactor core isolation cooling to remain on during the level reduction to aid level control is acceptable.

(4) The revision allowing actions to recover the water level prior to depressurization if the level drops below the minimum steam cooling reactor water level is acceptable because it avoids unnecessary depressurizations.

(5) Bypass of interlocks as necessary to prevent emergency coolant injection into the RPV during water level reduction is acceptable.

(6) The generic guidance regarding the main steam-line (MSL) and offgas high radiation interlock bypass to avoid MSL isolation is acceptable. When implementing plant-specific procedures, the licensees should perform an evaluation considering the following:

   (a) Additional steps to ensure that, following bypass of the radiation interlock, the operators confirm that offgas and support systems that will be relied upon during the non-isolation condition are available.

(b) For ATWS events which lead to severe core damage, the adequacy of maintaining the radiation interlock in a bypass condition should be considered within the accident management studies.

(c) Including consideration of (a) and (b), define radiological and plant conditions that require positive steps to isolate the containment. Also consider any measures that may be prudent to confirm that the ATWS isolation bypass condition is appropriate and has not resulted from misdiagnosis. Credit may be taken for remaining automatic isolation interlocks (e.g., high main steam flow and high steam temperature) to deal with events that are of isolation concern.

(7) For non-isolation ATWS events that do not result in automatic trip of the recirculation pumps, manual runback before tripping the recirculation pumps as proposed in the EPG change is acceptable.

The staff has concluded that early boron injection and a higher level control strategy is likely to reduce the overall ATWS risk. However, on the basis of the risk assessment presented in the evaluation, the staff has determined that, because of the small contribution of ATWS sequences to core damage, a 10 CFR 50.109 backfit requiring RPV level control at high level would not be justified. Therefore, we have reached the following additional conclusion:

(8) For BWRs designed to inject boron solution through a standpipe below the core, the staff encourages that procedural guidance be modified to target water level control at the TAF+5 ft level (Strategy A) or as high as is possible while maintaining the level 2 feet (0.61 m) below the feedwater sparger. However, the staff considers control at any level between minimum steam cooling water level and 2 feet (0.61 m) below the feedwater sparger to be acceptable. Therefore, the Susquehanna deviation from the EPGs with respect to target water level control is acceptable.

## 5.0 References


(1) Nuclear Regulatory Commission, "Safety Evaluation Report Accepting NEDO-32164 and NEDO-32047," February 5, 1994.

(2) General Electric Co., NEDO-32047, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability," February 1992.

(3) General Electric Co., NEDO-32164, "Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS," December 1992.

(4) General Electric Co., NEDO-31331, "Emergency Procedures Guidelines," Revision 4, March 1987.

(5) Boiling Water Reactor Owners' Group, Letter from L. A. England (BWROG) to M. Virgilio (NRC) "Submittal of Requested Emergency Procedure Guidelines Modifications Addressing ATWS/Stability Issue," March 21, 1994.

(6) Operations Engineering, Inc., OEI Document 9402-3, "The Management of ATWS by Boron Injection and Water Level Control (Preliminary Unverified)," March 1994.

(7) Operations Engineering, Inc., OEI Document 9402-3, "The Management of ATWS by Boron Injection and Water Level Control," Revision 1, June 1994.

(8) Nuclear Regulatory Commission, Minutes from NRC/BWROG Meeting to Discuss BWROG Positions on EPG Issues, August 9, 1994.

(9) Nuclear Regulatory Commission, "Draft Safety Evaluation Report on BWROG Proposed EPG Modifications," June 16, 1995.

(10) Boiling Water Reactor Owners' Group, BWROG 95078, "Request for Comment on Draft Safety Evaluation of Proposed Emergency Procedure Guidelines-Boiling Water Reactor Owners' Group (BWROG) Response," September 15, 1995.

(11) Jason W. Hartzell, "An Investigation of BWR Stability Following a MSIV Closure Initiated ATWS using TRAC-BF1," MS Thesis, Pennsylvania State University, 1992.

(12) P. Saha, et. al., "RAMONA-3B Calculations for Browns Ferry ATWS Study," NUREG/CR-4739 (BNL-NUREG-52021), 1987.

(13) Nuclear Regulatory Commission, "Regulatory Analysis Guidelines of the NRC," NUREG/BR-0058, Revision 2, 1995.

(14) Nuclear Regulatory Commission, "Analysis of Core Damage Frequency: Peach Bottom, Unit 2 Internal Events," NUREG/CR-4550, Volume 4, Revision 1, 1989.

## Appendix A

## Staff Evaluation of BWROG Response to Draft Safety Evaluation of Proposed EPG Modifications

In response to a draft of this safety evaluation, BWROG submitted its comments and additional information supporting boron remixing.[1] The information sent discussed four plant transients involving lower plenum thermal stratification and provided additional documentation of the 1981 remixing tests done by the General Electric Co. (GE). In its response, the BWROG concludes that:

(1) as water level is reduced from TAF+5 feet to MSCWL, reactor power level must decrease with the amount of decrease depending on the code used, the exposure, and the analytical assumptions, and

(2) the information sent to the staff regarding boron remixing adequately demonstrates that stratified boron will remix during an ATWS.

This appendix documents the staff's review and disposition of the BWROG comments.

1.0 <u>The Effect of Level Reduction on Core Thermal Power</u>

The staff has reviewed the effect of level reduction on thermal power and has used these results to estimate the suppression pool temperature as a function of time for an ATWS with standby liquid control (SLC) system failure. The following sources were considered:

(1) TER calculations (TRAC-BF1 and RAMONA-4B)
(2) NEDO-32164 (TRACG calculations)
(3) EPRI NP-5562 (TRACG with 1D Kinetics)
(4) PP&L SABRE Code[2]

The staff chose to include only calculations that are internally consistent in power, flow, and core void fraction over the range of conditions considered and that are based on well validated and widely accepted codes. RETRAN analyses were rejected in this low-flow regime because RETRAN uses a multiplicative slip model. CRAC was not considered because it predicts non-physical results (see TER

Appendix A). The SABRE results were adjusted to account for differences in assumptions for the first 4 minutes in the transient.

The data from these sources were used to produce Figure A.1 which plots the power as a function of collapsed level. With the exception of the TER results, the staff interpolated core power levels from published plots. For the NEDO-32164 results, the staff used the values for the vessel steam flow and, after accounting for condensation affects in the presence of cold feedwater, determined a reactor power level. This was necessary because the calculation involved an instability and the power level was impossible to determine accurately. Brookhaven National Lab performed additional calculations with RAMONA-4B and the staff completed numerous TRAC-BF1 sensitivity calculations that are not specifically reported here. All of these sensitivity studies show the same trend as the data plotted in Figure A.1.

Once the power vs. level data were gathered, the staff assumed a downcomer level as a function of time consistent with the TER results and calculated the suppression pool response. It is necessary to account for the heat generation during the initial phases of the transient because merely starting with the level at its target does not account for the initial pool temperature rise (refer to TER Figure 1). Without this initial rapid increase, the time to heat capacity temperature limit (HCTL) and margin at HCTL are incorrectly calculated. This error arises because the temperature rises at a different slope, depending on where the reactor level is, and longer times to HCTL tend to exaggerate the margin between the two strategies.

The results of the suppression pool calculation are presented in Table A.1. As can be seen, the TER results compare well with NEDO-32164 results; the SABRE results show an additional 5 minutes of margin; and EPRI NP-5562 results show an additional 3 minutes of margin. The conclusion that the staff draws from these results is that there remains a great deal of uncertainty in the calculated time to HCTL because of differences in codes and analytical assumptions. Furthermore, the additional time gained at low level is not significant enough to procedurally account for the low probability event of an SLC system failure due to the adverse impact on the higher probability SLC operable event. In other words, the staff concluded that the potential recovery actions that operators may take in the additional 3 to 6 minutes gained by low-level control will not significantly alter the overall risk.

## 2.0 Boron Remixing

In its DSER response, the BWROG also sent additional information about boron remixing. This information comes from four plant transients involving lower plenum thermal stratification and a 1981 GE experiment designed to investigate boron remixing. The test indicates very rapid remixing (approximately 2-3 minutes) at flow rates above 5-8% of rated flow; the plant data indicate that fast remixing (less than 10 minutes) only occurs with flow rates greater than 15 percent of rated flow. In the opinion of the staff, the BWROG's explanations of these discrepancies do not adequately account for these differences. The staff remains uncertain as to why the plant and experimental data differ as much as they do, but the difference is likely caused by several factors including, but not limited to, the following:

(1) poor instrumentation in the plant
(2) poor documentation
(3) differences in specific gravity
(4) different volumes to be remixed

Because of the uncertainty of boron remixing efficiency, the staff concludes that it is prudent to choose the level control strategy that allows the most time to remix before containment limits are reached. However, fast remixing is expected at the high natural recirculation flow rates (22-25% of rated) that will be achieved following the level increase after HSBW injection. Therefore, either level control strategy will provide adequate remixing.

3.0 <u>References</u>


(1) Boiling Water Reactor Owners' Group, BWROG 95078, "Request for Comment on Draft Safety Evaluation of Proposed Emergency Procedure Guidelines-Boiling Water Reactor Owners' Group (BWROG) Response," September 15, 1995.

(2) Pennsylvania Power and Light Co., Technical Report, NE-92-01, "Technical Basis for PP&L's Approach to ATWS Procedural Guidance," Pennsylvania Power and Light Company Technical Report, June 5, 1992.

**Table A.1** Margin to HCTL for different calculations

Results of SLCS Failure Suppression Pool Heatup Rates

| Code | Level | Time (min) to HCTL | Margin[5] (min) |
|---|---|---|---|
| TRAC-BF1[1] | TAF+5′ | 12.52 | 0.72 |
| TRAC-BF1[1] | MSCWL | 13.24 | |
| RAMONA[1] | TAF+5′ | 8.8 | 1.9 |
| RAMONA[1] | MSCWL | 10.7 | |
| TRACG[2] | TAF+5′ | 13.2 | 5.3 |
| TRACG[2] | MSCWL | 18.5 | |
| TRACG[3] | TAF+5′ | 8.7 | 2.3 |
| TRACG[3] | MSCWL | 10.0 | |
| SABRE[4] | TAF+5′ | 13.6[6] | 5.6 |
| SABRE[4] | MSCWL | 19.2[6] | |

[1]Staff calculations

[2]EPRI NP-5562 (1987) results; 40% steam condensation assumed; TRACG with 1-D kinetics

[3]GE NEDO-32164 (1992) steam flow results; level ramp is the same as that used for NP-5562 results

[4]SABRE results; TAF+5 case from PP&L Technical Report NE-92-01 and MSCWL results from response to staff RAI; level ramp is the same as that used for NP-5562 results

[5]Represents the difference between the time to HCTL for the two levels in the table

[6]Letter from J. March-Leuba (ORNL) to T. Ulses (NRC), "More accurate data for Draft SER on ATWS/EPG," October 26, 1995

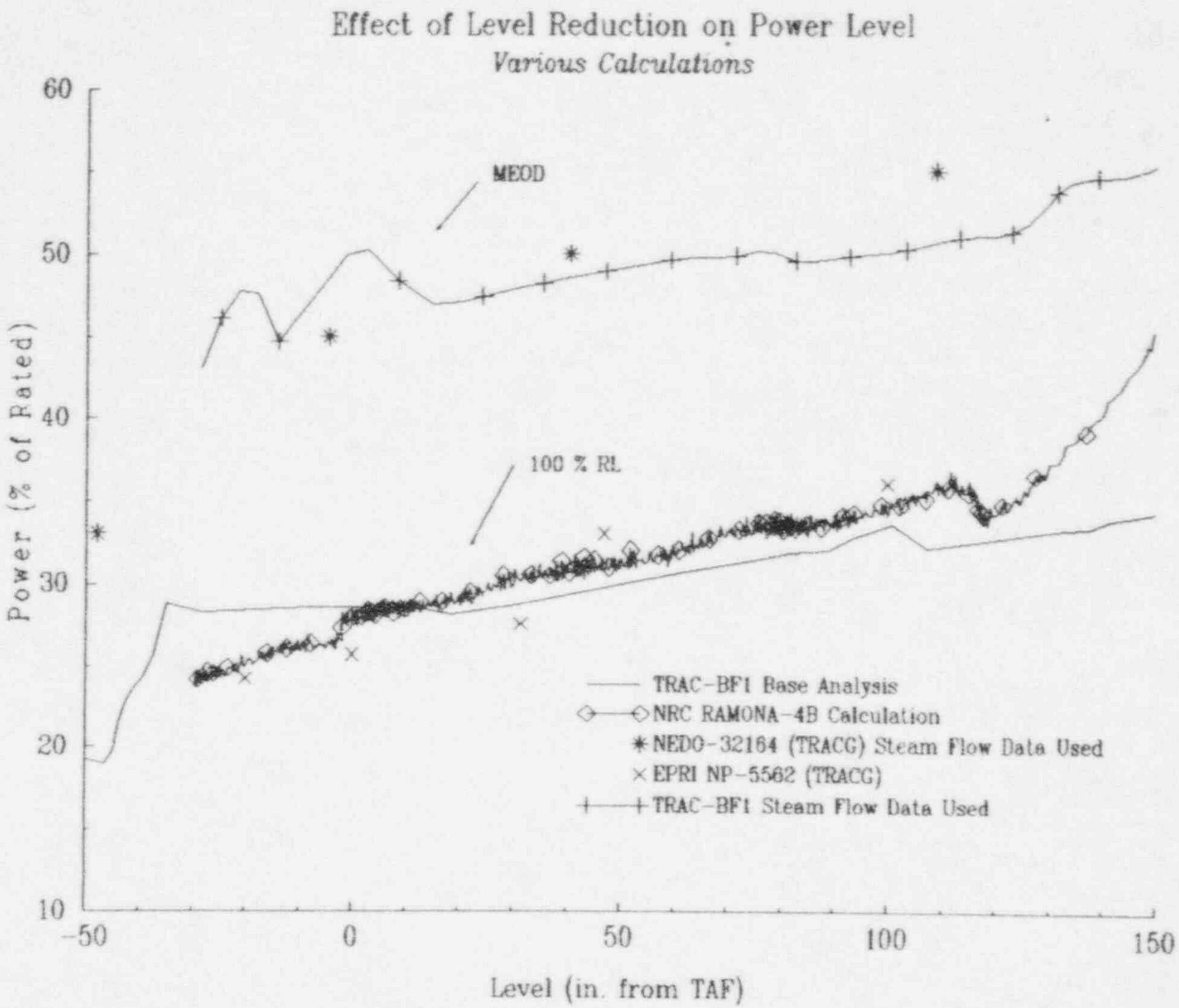


**Figure A.1** Effect of Collapsed Level on Core Thermal Power