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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 19, 1996

Mr. George A. Hunger, Jr.
Director-Licensing, MC 62A-1
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
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SUBJECT: STAFF REVIEW OF MODIFICATIONS TO REVISION 4 OF THE BOILING-WATER REACTOR (BWR) EMERGENCY PROCEDURE GUIDELINES - LIMERICK GENERATING STATION, UNITS 1 & 2 AND PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 & 3

Dear Mr. Hunger:

The staff has issued its safety evaluation (SE) on the recent BWROG-proposed modifications to the BWR Emergency Procedure Guidelines. The staff is providing this information to ensure that licensees are aware of the conclusions of the staff's review. Both the staff and the Advisory Committee for Reactor Safeguards (ACRS) agree that, for BWRs injecting standby liquid control through a standpipe below the core, maintenance of level above top-of-active fuel (TAF) is the superior water control strategy in an anticipated transient without scram (ATWS) event. The staff recommends a level around TAF +5 feet (1.52 m), or as high as possible while still maintaining the level at least 2 feet (0.61 m) below the feedwater sparger. Although control at any level between the minimum steam cooling water level and 2 feet below the feedwater sparger was found to be acceptable, both the staff and ACRS urge that a high-water-level control strategy be adopted. Additional details are provided in the enclosed SE.

You should also note the staff's position on bypassing the Main Steam Isolation Valve (MSIV) high radiation closure interlock during ATWS. The staff agrees with the BWROG's qualitative arguments that keeping the MSIVs open significantly reduces containment loading and makes level control much simpler. However, the acceptability of this change is conditional on a plant-specific evaluation by each licensee to assure that, in the event of gross

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fuel failures, consideration has been given to such items as equipment accessibility, potential off-site radiological doses, and the appropriate time to manually close the MSIVs.

Sincerely,

/S/

Frank Rinaldi, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

/S/

Joseph W. Shea, Project Manager
Project Directorate I-2
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Docket Nos. 50-352/50-353
and 50-277/50-278

Enclosure: Safety Evaluation Report

cc w/encl: See next page

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fuel failures, consideration has been given to such items as equipment accessibility, potential off-site radiological doses, and the appropriate time to manually close the MSIVs.

Sincerely,

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Docket Nos. 50-352/50-353
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Enclosure: Safety Evaluation Report

cc w/encl: See next page

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UNITED STATES
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WASHINGTON, D.C. 20555-0001

June 6, 1996

Kevin P. Donovan
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SUBJECT: ACCEPTANCE OF PROPOSED MODIFICATIONS TO THE BOILING WATER REACTOR (BWR) EMERGENCY PROCEDURE GUIDELINES (TAC NOS. M89489 AND M89629)

Dear Mr. Donovan:

The staff has completed its review of: (1) your submittal transmitted by letter (BWROG-94038) dated March 21, 1994, requesting our approval of proposed modifications to the BWR Emergency Procedure Guidelines (EPGs) (NEDO-31331) to address reactor core instabilities; and (2) the Operations Engineering, Inc., (OEI) Document 9402-3, "The Management of ATWS by Boron Injection and Water Level Control," submitted by letter (BWROG-94111) dated September 16, 1994. OEI Document 9402-3 gives the analyses used by BWROG to justify its ATWS control strategy.. The staff has also reviewed the BWROG submittal transmitted by letter (BWROG-95078) dated September 15, 1995, responding to the staff's request for comments on a draft safety evaluation report (SER). The draft SER was placed in the Federal Register for comment and copies were transmitted with an invitation for comment to other interested parties. The final SER, enclosed, reflects modifications made in response to BWROG and ACRS comments (no other comments were received) and defines the basis for the staff's acceptance of the proposed EPG modifications.

Both the staff and the ACRS agree that, for BWRs injecting standby liquid control through a standpipe below the core, maintenance of level above top-of-active fuel (TAF) is the superior water level control strategy in an anticipated transient without scram (ATWS) event. The staff recommends a level around TAF +5 feet (1.52 m), or as high as possible, while still maintaining the level at least 2 feet (0.61 m) below the feedwater sparger. Although control at any level between the minimum steam cooling water level and 2 feet below the feedwater sparger was found to be acceptable, both the staff and ACRS urge that a high-water-level control strategy be adopted. Additional details are contained in the enclosed SER.

The staff is sending a copy of this letter and the enclosed SER to all BWR licensees and understands that all BWR licensees will revise their emergency operating procedures consistent with the approved modifications. The suggested implementation program contained in Enclosure 1 to the staff's September 12, 1988, letter to the BWROG transmitting the SER on NEDO-31331, Revision 4 remains applicable. The staff requests that the BWROG monitor the progress of implementation and inform the NRC of the number of plants completing implementation and the number selecting each water level option.

The staff does not intend to repeat its review of the matters found acceptable, as described in the enclosed SER, on the proposed modifications for NEDO-31331, except to ensure that the material presented is applicable to the specific plant involved.

In accordance with procedures established in NUREG-0390, "Topical Report Review Status," the staff requests that the BWROG publish an accepted revision of this report within three months of receipt of this letter. The accepted revision should incorporate this letter and the enclosed Safety Evaluation, including the attached technical evaluation report. The accepted revision should include an -A (designating accepted) following the report identification designation.

Sincerely,

David B. Matthews, Chief
Generic Issues and Environmental
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Office of Nuclear Reactor Regulation

Project No. 691

Enclosure: As stated

cc: see attached list

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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 References

(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

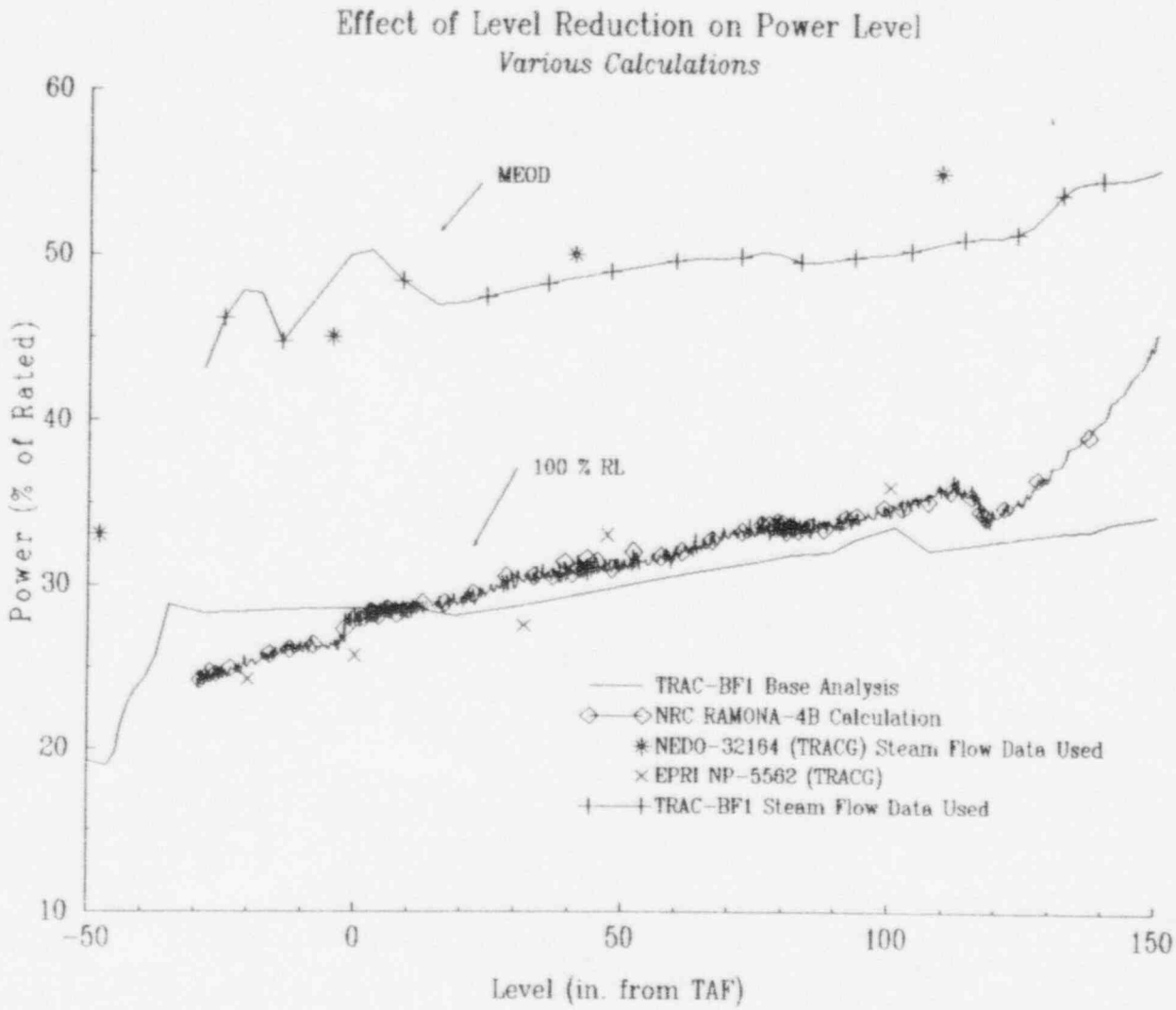

Figure A.1 Effect of Collapsed Level on Core Thermal Power

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SUMMARY

This technical evaluation report documents our review of changes to the emergency procedure guidelines (EPG's) proposed[1] by the Boiling Water Reactor Owners' Group (BWROG). The main purpose of these EPG changes was to minimize the likelihood of large power oscillations during anticipated transients without scram (ATWS), which could pose a threat to fuel integrity.[2] During the review of these changes, the Nuclear Regulatory Commission (NRC) and the Advisory Committee for Reactor Safeguards (ACRS) have raised a number of additional issues, which have prompted an overall review of the ATWS management strategy in addition to the proposed changes to the EPG's. This review has concentrated mainly on five technical issues

(1) Large-amplitude unstable power oscillations, and their impact on core coolability.

(2) Ultimate suppression pool temperature using different level control strategies.

(3) Core coolability with downcomer water level below top of active fuel.

(4) Pressure oscillation induced by SRV cycling, low downcomer water level, or condensation effects produced by cold feedwater injection.

(5) Applicability of 1/6 scale boron destratification experiments to a full-scale reactor.

The main conclusions from our review of the above issues are:

(1) Lowering water level below the feedwater spargers to reduce core-inlet subcooling is an effective action to mitigate the consequences of large power oscillations. This action reduces core flow and inlet subcooling by preheating cold feedwater with vessel steam, which reduces power and the likelihood of instabilities. In addition, the suppression pool heat load is also reduced by up to 40% because of the steam that is condensed in the downcomer. For this action to be effective against power oscillations and to minimize the suppression pool heat load, the water level reduction must be initiated immediately upon ATWS confirmation.

(2) The results of different ATWS calculations appear to be fairly sensitive to modeling assumptions. Nevertheless all "credible" calculations indicate that higher water levels (e.g., top of active fuel plus five feet) result in higher margin to heat capacity temperature limits (HCTL) than lower levels (e.g., minimum steam cooling water level) when the standby liquid control (SLC) system works. Staff calculations show a small advantage (of the order of 1 minute) for the low-level strategies when SLC fails. These calculations also show that under SLC failure assumptions, both strategies reach the HCTL within approximately 10 minutes.

(3) As part of this review, NRC has performed a number of ATWS simulations with best-estimate computer codes. The results of these calculations indicate that core coolability is not compromised as long the downcomer water level is maintained above the minimum steam cooling water level (MSCWL). Experimental data from the TLTA test confirm these analyses.

(4) Analyses indicate that pressure oscillations induced by either SRV cycling, low downcomer water level, or condensation effects produced by cold feedwater injection are relatively small and do not compromise core coolability or water level measurement instrumentation.

(5) The success of the modified-EPG strategy on reactors with stand-pipe SLC injection relies on remixing the stratified boron in the vessel bottom by raising water level. A remixing test,[10] which was performed in a 1/6 scale facility, showed that boron destratification should occur with recirculation flows greater than ~10%. Our review of this test has shown that proper scaling laws were used to extrapolate the test results to full scale, and cold-coolant stratification events in operating reactors confirm the general results from the tests. Both the tests and operational events indicate that stratified boron should remix if the recirculation flow is increased to at least 15%. However, the review of this operational data has revealed significant uncertainty with respect to the time constant for destratification, which may be longer than predicted by the scaled experiments. Our review of the available data indicates that remixing should take place in less than 10 minutes if the recirculation flow is increased to at least 15%.

(6) Even allowing for a 10-minute remixing time constant, best estimate staff calculations indicate that for all strategies level control between MSCWL and 2 feet below the feedwater sparger results in a success path.

(7) Reduction of HSBW from the equivalent of 755 ppm (reference to hot) in the Rev. 4 EPGs to 355 ppm (reference to cold, or 487 ppm reference to hot) in the modified EPGs by eliminating unneeded conservatisms is an acceptable way to minimize the possibility of emergency depressurization during ATWS events.

(8) Water level instrumentation inaccuracies induced by large power and pressure oscillations are not expected to impact significantly the management of ATWS events because the modified EPGs reduce promptly the water level below the feedwater spargers to eliminate the oscillations.

(9) Allowing the recirculation pumps to run back in a controlled fashion before they are tripped manually is an acceptable action if the reactor is not isolated, and it should reduce the probability of unnecessary isolations.

(10) We do not have sufficient data to determine whether bypassing the MSL and off-gas high radiation interlocks is an acceptable action to maintain the main condenser as a heat sink when evidence of gross fuel failure exists. The acceptance of this action from a technical basis would require a plant- and procedure-specific evaluation.

INTRODUCTION

Proposed EPG Modifications

The modified EPGs provide two main modifications to the Rev.4 EPGs:

(1) To mitigate the consequences of large power oscillations that may be caused by instabilities, the modified EPGs require the operator to lower the water level below the feedwater spargers immediately after an ATWS event is recognized. In addition, boron injection is now mandated if 25% or larger oscillations are observed. The purpose of these modifications is to allow cold feedwater to mix with vessel steam in order to reduce subcooling and the probability of power oscillations.

(2) To reduce the integrated heat load to containment during ATWS events, the modified EPGs reduce the recommended hot shutdown boron weight (HSBW) by eliminating some of the conservatisms in the calculation. The purpose of this modification is to minimize the amount of time the reactor must operate at the flow-stagnation power level before it is shut down by raising the water level. The new HSBW calculation procedure is defined in Appendix A of the EPGs, but this review assumes that the reduced HSBW is an integral and necessary part of the proposed modifications. The proposed HSBW reduces the boron injection time by ~35%. This shorter time makes the differences between alternate level control strategies less significant.

Additional modifications to the EPGs include:

(1) Main steam line (MSL) and off-gas high radiation interlocks are bypassed to maintain the main condenser as a heat sink even if there is evidence of gross fuel failure.

(2) The ATWS rule requires that the recirculation pumps be tripped automatically upon detection of an ATWS event. The EPGs also require manual recirculation pump trip to supplement the automatic trip. In the modified EPGs, the recirculation pumps are allowed to run back before they are tripped manually to avoid the possibility of unnecessary level isolations.

(3) The operator may (but is not required to) control the water level as low as the minimum steam cooling water level (MSCWL), even when the suppression pool is not heating up. In Rev. 4, the minimum control range for these conditions was top of active fuel (TAF). This modification results in a single water level control range for all ATWS events. It must be noted that the modified EPGs do not require the operator to continue to lower the water level to the MSCWL if: (a) the reactor is shutdown by boron before the water level is completely lowered (because of inside-the-shroud injection, or an existing partial rod insertion), or (b) if all safety relief valves (SRVs) close at a higher water level (e.g., in a turbine trip with bypass ATWS event) so that containment integrity is not threatened.

LARGE-AMPLITUDE UNSTABLE POWER OSCILLATIONS

As a result of the LaSalle instability event, a number of studies were performed to predict the growth rate of the power oscillations should the reactor had failed to scram. As a result of these studies, it was concluded that the oscillations may have grown very large and may have compromised clad integrity.[2] The root cause of this oscillation amplitude growth was determined to be the large subcooling reached when feedwater heating is lost following turbine isolation. To solve this problem, the BWROG modified the EPGs to require immediate water level reduction to at least 2 ft below the

feedwater sparger. This water level reduction exposes the cold feedwater to a steam environment, which preheats it so that the core inlet coolant is close to saturation conditions.

Immediate lowering of the downcomer water level is the best action possible, not only to mitigate the consequences of large power oscillations, but also because it significantly reduces the heat load to containment by reducing the power generated in the core because of subcooling effects. Analyses[2] performed by the BWROG using TRAC-G clearly indicate that the large amplitude power oscillations are mitigated by a prompt reduction of water level below the feedwater sparger. These analyses are also confirmed by frequency domain stability calculations[3] which show a clear stabilizing trend induced by the power reduction which results from the increase in core inlet temperature.

WATER LEVEL CONTROL STRATEGY

The ATWS management strategy is a compromise between two conflicting approaches to minimize heat load to containment. On one hand, it is desirable to rapidly reduce reactor power in order to minimize the suppression pool heatup rate; on the other hand, it is desirable to inject and mix boron with the core coolant as fast as possible to permanently shutdown the reactor. The most effective action to minimize containment heat load is to lower the reactor vessel water level to reduce flow and reactor power. The conflict arises from the fact that reducing the recirculation flow also reduces boron mixing efficiency. In fact, for recirculation flows of approximately 5%, mixing is so poor that most of the sodium pentaborate solution injected by the SLC is not entrained by the coolant flow and stratifies in the bottom of the lower plenum (for plants with stand-pipe SLC injection). The EPGs reconcile these two approaches by three steps:

(1) The first step is to minimize the power early into the transient by reducing the water level to its minimum allowable. The minimum power that can be reached by lowering water level is the flow-stagnation power, which is reached when the external recirculation loop (through the jet pumps) is broken. Once this flow-stagnation condition occurs, further water level reductions do not reduce the power significantly. Flow stagnation may not occur even at MSCWL if the reactor was operating at high power. Under these conditions the power is greater than the flow-stagnation power until sufficient boron is entrained in the core coolant and the flow stagnates because of the lower core void fraction.

(2) Step two consists on holding this reduced power until sufficient boron has been injected to shutdown the reactor. This amount of boron is called the Hot Shutdown Boron Weight (HSBW), and it is calculated in a conservative fashion to cover all expected conditions.

(3) The final step is to raise the water level to the low-level scram setpoint in order to increase the flow and remix the stratified boron. This step shuts the reactor down, but the EPGs require continued boron injection until cold shutdown conditions are reached.

A fault with the EPG strategy is that the HSBW must be defined in a conservative fashion to cover all expected ATWS conditions. This conservatism forces the operator to hold the flow-stagnation power for a longer period than is required for all ATWS events except the limiting one; thus, the integrated

containment heat load is higher than a more effective shutdown would require. In essence, the EPG strategy produces the limiting ATWS containment heat load for most ATWS events.

Because of this inherent non-optimality of the EPGs, several different strategies have been proposed to optimize the management of ATWS events by boron injection and water level reduction. In the final evaluation, only two strategies have been compared: (1) Strategy B, the water level control strategy described above and currently implemented in the Rev. 4 EPGs, and (2) Strategy A, which was proposed in ref. 5 and consists of maintaining the water level higher than TAF (typically TAF + 5 feet), so that the flow stagnation occurs later into the ATWS event and a higher concentration of boron can be injected before the flow stagnation occurs.

A number of calculations[4-8] have been performed to determine the effect of controlling at different water levels on the ultimate suppression pool temperature. The results among different calculations vary significantly and indicate a large sensitivity to modeling assumptions. For this reason, NRC has performed a number of calculations for consistent comparison of the two strategies using two well-established codes (TRAC-BF1, and RAMONA-4B). Figure 1 summarizes one set of these staff calculations. This figure shows the suppression pool temperature following a MSIV isolation ATWS event using TRAC-BF1.

As it can be observed in Fig. 1, the staff calculations indicate that the higher-level strategy (TAF + 5') results in a lower suppression pool temperature than the low-level (MSCWL) strategy. This calculation also indicates that, as long as the hot shutdown boron weight (HSBW) is injected in approximately 20 minutes and the stratified boron is remixed promptly when the level is raised, the difference between the two level-control strategies is not significant in the sense that both strategies have sufficient margin to the heat capacity temperature limit (HCTL) and result in safe shutdown of the reactor. The results shown in Figure 1 also indicate that, if boron injection fails, both strategies reach the HCTL in 10 to 15 minutes and the difference in performance between the two strategies is not large.

Figure 2 shows the core power calculated by TRAC-BF1 for the same MSIV isolation ATWS events represented in Figure 1, and Figure 3 shows the core boron concentration for the two strategies. These two figures show more clearly the advantage of the higher-level strategy. As a consequence of maintaining the high level, the power generation is higher for the first 500 seconds, but after that the power drops significantly below the MSCWL case, because boron continues to be mixed for 750 seconds until a concentration of 300 ppm is reached. This boron concentration is sufficient to reduce the power and drop the flow to 5% (the flow at which boron is assumed not to mix), as seen in Figure 4.

The results of the staff calculations also indicate the need for immediate water level reduction once an isolation ATWS event is identified. As seen in Figure 1, the suppression pool is heating up very rapidly for the first few seconds when the level and power generation is high; thus, relatively small changes in assumptions for timing of operator actions in this phase of the transient result in very large discrepancies in the final suppression pool temperature. For example, the calculations reported in ref. 7 indicate that if the operator does not take immediate control of feedwater flow and other injection sources, the suppression pool reaches close to 150°F before the level is lowered and the pool heatup rate is reduced. Staff analyses based on the calculations of ref. 7 indicate that if the EPG procedures (which require prompt operator action) had been followed the final pool temperatures calculated by SABRE for the two level strategies would have been similar to the ones shown in Figure 1.

Figure 5 shows the core average void fraction for the two level-control strategies studied. This figure shows that as expected, to maintain the level at MSCWL, the reactor requires a higher void fraction than when the level is higher at TAF + 5'; thus, a higher power is required at the lower level to maintain the same flow (approximately 5%).

An unexpected result of the staff calculations is the small dependency observed between power and level once the level is lowered below TAF + 5'. Figure 6 shows the core thermal power as function of downcomer water level. This figure has been produced by plotting the power and level during a TRAC-BF1 and a RAMONA-4B calculation in which the feedwater flow is slowly ramped down to zero. We can observe that both codes, TRAC-BF1 and RAMONA-4B, predict that the dependency between power and water level is fairly flat around the TAF area. This is the reason why the differences between the two level strategies are not as large as expected when the SLC system fails.

The sharp step change in power seen in the TRAC-BF1 calculation shown in Figure 6 at around TAF - 40" corresponds to the stagnation of the external flow recirculation path. The RAMONA-4B calculation (which does not correspond to the same reactor or initial conditions that the TRAC-BF1 calculation) predicts slightly higher recirculation flows than TRAC-BF1 and does not predict flow stagnation even for a water level as low as the top of the jet pumps (TAF -50"). For levels below TAF - 40", the core inlet flow is greatly reduced and equals only the flow of steam leaving the core; this results in a significantly lower power. The value of the flow-stagnation power calculated by TRAC-BF1 is slightly less than 20%. This number was previously assumed to be of the order of 8% based on a extrapolation of a number of REDY analyses.[10] The larger value of the flow-stagnation power is consistent with other BWROG calculations[2] and is attributed to the fact that REDY uses a multiplicative slip void model, which is incapable of predicting accurately the void fraction distribution under flow-stagnation conditions.

The results presented in Figs 1 through 5 are based on TRAC-BF1 calculations. As mentioned above, the staff also performed a number of RAMONA-4B calculations to evaluate the sensitivity to different operating conditions (i.e., reactor and input deck) and modelling assumptions (i.e., code used). Figures 7 and 8 show a comparison of suppression pool temperatures calculated by both codes for the case of TAF + 5' and MSCWL, respectively. We observe that the calculated pool temperatures are in fairly good agreement, and that both codes predict similar trends. Overall, RAMONA-4B with the particular input deck used tends to predict higher flows than the TRAC-BF1 input deck used in these analyses; this higher flow results in higher power and higher suppression pool temperature. However, the differences between the two codes are judged not to affect significantly the main conclusions from this report.

CORE COOLABILITY WITH WATER LEVEL BELOW TAF

Because the proposed EPG modifications allow the water level to be controlled as low as the MSCWL (which is approximately TAF minus 30 inches), some concerns have been raised about core coolability, especially under low power conditions where the void fraction may be low. The main concern is that the two-phase level may drop below the top of the core and the clad may overheat when the collapsed water level is dropped bellow the top of active fuel.

Our review of the experimental data from the TLTA test facility[9] has shown that fuel bundles are fully covered with two-phase mixture when the collapsed downcomer level is at the MSCWL even under 6% decay heat conditions.

The NRC calculations have also shown that the two-phase level is well-above the core (indeed, in most cases, the two-phase level is at the separator level) even when the level is at the MSCWL. As seen in Figure 5, the core average void fraction is lower with the level at MSCWL than at nominal conditions (time zero in Figure 5). This fact indicates that the core is likely to be covered with a two-phase mixture. The core coolability is confirmed by studying the peak clad temperatures calculated by TRAC-BF1. Figure 9 represents a composite of ten different TRAC-BF1 calculations, and it shows the maximum clad temperature in the low-power channel (expected to have the limiting performance). This set of TRAC-BF1 calculations includes collapsed water levels from TAF + 5 to MSCWL, and it shows that the clad temperature is well within acceptable limits. Indeed, for most transients, the peak clad temperature is lower than at rated conditions (time zero).

Figure 9 shows only the low-power peripheral channel for ten different TRAC-BF1 runs. Figure 10, in contrast, shows a composite of all the peak clad temperatures calculated by TRAC-BF1 for all channels in the two base runs: MSCWL and TAF + 5' level control with boron injection. This figure confirms that the peak clad temperature is within acceptable bounds, and it indicates that the two-phase level is well above the top of active fuel even when the collapsed water level is at the MSCWL.

PRESSURE OSCILLATIONS

Large, damaging pressure oscillations have been postulated during ATWS events by three different mechanisms: (1) SRV cycling, (2) level oscillations when collapsed water level drops below top of active fuel (where the downcomer flow area is reduced significantly), and (3) induced by condensation of cold feedwater in steam space when the water level is lowered below the spargers.

The first two concerns were raised mainly in a Pennsylvania Power and Light (PP&L) report,[6] where large power oscillations were observed by the SABRE code when the collapsed water level was lowered below the top of active fuel. Since those calculations in 1992, the models in the SABRE code has been upgraded to represent more closely the physical processes in a BWR. Calculations using the new SABRE models[7] do not show the large power oscillations that were observed in the 1992 report.[6] Two main modifications to the SABRE code affect these results: (1) the SRV banks are opened at staggered pressure setpoints - in the 1992 version, SABRE would open all SRV's when the pressure reached the setpoint. (2) The SABRE models have been upgraded to model more accurately reverse flow at the channel inlet - the 1992 model would result in a large reactivity transient following channel-inlet reverse flow caused by a large ingress of subcooled lower-plenum coolant.

In summary, concerns (1) and (2) have been eliminated by the new set of calculations performed in ref. 7. NRC has performed independent calculations using TRAC-BF1 and RAMONA-4B. These calculations included collapsed water level below TAF and significant SRV cycling, and did not exhibit any of the large power oscillations reported in the PP&L report. We conclude, thus, that those large power oscillations were most likely an artifact of the old SABRE models (which have been corrected) and are not inherent to low-level ATWS management strategies.

Condensation-induced pressure fluctuations (concern (3) above) have been addressed in part by the TRAC-BF1 and RAMONA-4B calculations. Both of these codes have a mechanistic model for heat transfer between the subcooled liquid and steam phases; therefore, they compute the average equilibrium pressure that is reached in the node where steam condensation takes place. The result of these calculations exhibit pressure and power oscillations that are correlated with SRV cycling, but the magnitude of these oscillations is acceptable (see Figure 2), and it only results on peak clad temperature oscillations of the order of 5 °C, as seen in Figs 9 and 10.

A simple enthalpy balance in the vessel indicates that cold feedwater condenses no more than 40% of the steam produced in the core. The only physical mechanism available to relieve the remaining 60% of the steam is by lifting the SRV valves, which set the minimum and maximum operating pressure; thus, the pressure (other than possibly very high-frequency, low-energy components) must remain close to the SRV setpoint.

BORON MIXING

Boron mixing issues can be classified in two categories:

(1) Mixing or entrainment of boron injected through the standing pipes in the lower plenum.

(2) Remixing or destratification by increasing flow after sufficient HSBW has been injected and has stratified in the bottom of the lower plenum.

These two categories are discussed separately.

Boron Mixing (Entrainment) Correlation

The present reevaluation of ATWS management strategies was prompted in part by the availability of new boron mixing data as reported in ref. 5. The correlation used by the BWROG is shown in Fig. 11. This correlation assumed conservatively that the mixing efficiency would diminish for flows lower than 25% and would become zero for flows below 5%. The new experimental data is based on a full scale mockup and was reported in ref. 3. The new data suggests that mixing is almost perfect (i.e., 100% efficiency) even for flows as low as 4% to 5%.

The BWROG has performed a series of simulations to study the effect of the improved mixing efficiency. Their conclusion is that this improvement in efficiency helps, but the final results are not qualitatively different because of the flow-stagnation problem. As long as the recirculation flow stagnates, boron mixing ceases and it stratifies in the bottom of the vessel. Staff calculations confirm this trend: the success or failure of an ATWS event is controlled mostly by the time it takes to inject the HSBW while the reactor is at flow-stagnation power. Increases in boron efficiency only increase marginally the amount of boron mixed in the core coolant before the flow stagnates. For example in the staff calculations shown in Figs. 1-6, the boron concentration achieved before recirculation flow stagnation in the MSCWL case is ~100 ppm (hot), which is only 20% of the concentration required for hot shutdown (500 ppm hot).

Although significant changes in mixing efficiency as a function of core flow may affect which water level results in the lowest containment temperature, we conclude that the efficiency changes indicated by the new available experimental data do not impact significantly the choice of ATWS management strategy.

Boron Remixing (Destratification) Correlation

Successful ATWS management with all proposed strategies requires raising water level to remix the cold sodium pentaborate solution that precipitated to the bottom of the vessel when flow-stagnation conditions were reached. Figure 12 shows the remixing correlation determined from the 1/6 scale boron remixing experiments.[10] The remixing efficiency is defined in terms of a "mixing time constant" (the time it takes to remix 50% of the stratified boron solution). From these experiments it was concluded that boron remixing would be essentially instantaneous (approximately 10 s) if more than ~10% recirculation flow is established once the level is raised.

Our review of the scaling methodology of the 1/6 scale facility results to full reactor conditions indicates that the scaling laws used are acceptable. The mass flow rates and times to remix were scaled according to the modified Froude number (1/Richardson number), which scales the density and buoyancy terms. The Reynolds number in the facility is ~100 times smaller than in the reactor, which makes the results conservative from the point of view of friction. The Peclet number is very large for both the facility and the reactor, indicating that conductive heat transfer is not a principal mechanism for destratification. Thus, we conclude that the Froude number scaling is adequate for these test.

The BWROG has presented to NRC and ACRS data from some operational transients where lower plenum stratification was observed. These data include events from one foreign and two U.S. reactors. These data confirm the scaled experiments and indicate that the lower plenum destratifies once flow greater than 15% gets established. The data quality, however, is not adequate to evaluate the timing of these transients in detail (typically the data is stored in stripcharts with a 1 inch per hour time scale and may have "pen offsets"), but trends can be observed in the data which seem to indicate that full destratification of the lower plenum is accomplished in 10 min or less from the time that significant (> 15%) recirculation flow gets established. This time to remix contradicts the 10 s predicted by the 1/6 scale facility, but is sufficient to guarantee a success path for both level strategies according to the best estimate staff calculations documented in this TER.

In summary, the results of the scaled facility appear to be properly scaled and should be applicable to the full-size reactor conditions. Data from operational events raise some doubt about the remixing time constant that was scaled from the experiments, but confirm that: (1) the stratified cold boron solution will remix if the recirculation flow is increased to at least 15%, and (2) the remixing time should be less than 10 minutes.

Impact of Uncertainty on Boron Remixing Time Constant

The staff analyses presented earlier in this report indicate that the differences among level-control strategies is not significant as long as the boron destratification occurs within a time scale as predicted by the 1/6 scale experiments. In this section, we present results of analyses that assume that boron

does not destratify when the water level is raised once the HSBW is injected. These results indicate that the higher level strategies have a significant advantage over the lower levels, and they have increased margin to the time when HCTL is reached.

Figure 13 shows the core thermal power for the two level-control strategies if we assume that boron is not destratified when the level is raised. It can be observed from this figure that the power for the lower level strategy (MSCWL) is significantly higher than the power for the higher level strategy. This is due to the fact that the lower-level strategy has a smaller concentration of boron before the SLC injection begins to stratify; therefore, when the level is increased the flow increases and the power is higher than with the higher dissolved boron concentration of the higher-level strategy.

Figure 14 shows the flows reached after level increase (with failure to destratify) for both strategies. As it can be observed, both strategies reached approximately the same recirculation flow (greater than 20%) even though the power generation is quite different (see Fig. 13). This fact is not unusual and it indicates that the "natural circulation line" in the power to flow map is almost vertical, as expected. The calculations of Fig. 14 indicate that the reactor achieves a flow greater than 20% when the level is raised to remix the stratified boron; all available data indicates that stratified boron will remix at this high flow.

Figure 15 shows a comparison of suppression pool temperatures for the two level-control strategies when we assume that boron does not destratify and no additional boron is injected once the water level is raised. As seen in this figure, the higher-level strategy results in lower pool temperatures and more time before it reaches the HCTL (400 s for the MSCWL strategy versus 1400 s for the TAF + 5' strategy).

The calculations in Fig. 15 are extremely conservative; they not only assume that boron does not destratify, but also assume that no additional boron is injected once the level is raised. Figure 16 shows a slightly less conservative calculation, where the stratified boron is assumed not to remix, but we take credit for the additional boron that is injected by the SLC system after the level is raised and the flow increases back to 20%. When this additional boron injection is accounted for, the time available before HCTL is reached is increased significantly, and it provides more than the 10 minutes that may be required to remix the stratified boron.

In the above calculations, the staff has assumed prompt response by the operators, which are assumed to act within the first two minutes to reduce the water level and start boron injection. The most important action to take to minimize power is to reduce the water level, but prompt boron injection is also required for a successful completion of the transient. If boron is not injected promptly, the time to inject the HSBW is delayed accordingly, and the integrated heat load to containment increases.

OTHER REVIEW ISSUES

Hot Shutdown Boron Weight

The success of the ATWS strategy is directly linked to the assumption used to determine the time required to inject the HSBW. Adding too much conservatism in the HSBW calculation is actually detrimental to safety and may require emergency depressurization of a critical reactor. The HSBW is

calculated based on a conservative calculation that includes: a typical core in the most reactive condition, rods at the rod-block withdrawal limit, equilibrium xenon, and no voids. This calculation was performed and resulted in a required boron weight equivalent to a boron concentration of 355 ppm referenced to the density of light water at cold shutdown conditions. When the changes of water density as a function of operating temperature are taken into account, this minimum HSBW-equivalent concentration is ~487 ppm. In both cases (355 ppm cold, or 487 ppm hot), the weight of boron inside the vessel is the same; for a typical BWR/4 it is ~348 lb_m.

The Rev.4 EPGs HSBW calculation procedure was equal to the modified procedure except for the xenon concentration. Rev.4 required no xenon, and the modified EPGs call for equilibrium xenon. In addition, the new HSBW calculation takes advantage of the requirement that rods be located at withdraw limits, while the old Rev. 4 calculations were performed at a conservative "all rods out" condition to simplify the calculations. The old Rev. 4 calculated HSBW was equivalent to a hot concentration of 755 ppm (equivalent to ~560 lb_m of boron in a typical BWR/4). Thus, the modified EPGs propose to reduce the HSBW by approximately 35%.

The removal of unnecessary conservatism is an acceptable action because of the detrimental effect of the additional time the reactor must wait at the flow-stagnation power when the conservatism is maintained. This is a case where a conservatism in the analyses makes the actual transient worse. In the unlikely case that recriticality occurs after the level is raised, the consequences to the core are acceptable and the EPGs require lowering the water level and injecting additional boron solution. The HSBW is calculated assuming the total volume of water in the vessel (with level at high-scram setpoint) and the recirculation loop. Even if the HSBW is miscalculated, when the level is dropped again by procedure, the concentration increases and the reactor is likely to become subcritical at that lower level. Thus, underestimation of boron injection requirements in the HSBW calculations does not result in unacceptable consequences.

Water Level Instrumentation

Water level instrumentation is crucial during ATWS events, because it is the most important instrument that the operator relies on to follow the EPGs. The Electric Power Research Institute has performed a series of tests and analyses[11] to verify that this instrumentation responds adequately during ATWS events in the presence of large power oscillations. The concern here is that the power oscillations produce: (1) actual water level oscillations, and (2) pressure pulses that when sensed at different times in the two sides of the differential pressure sensors result in observed, but not real, level oscillations.

The main results of these analyses are documented in ref. 11. These results indicate that during ATWS event with large power oscillations, we should expect real level oscillations as large as 0.2 m (0.6 ft), and sensed level oscillations as large as 0.3 m (1 ft). The expected pressure oscillations at the two sides of differential pressure transducers are expected to be larger than the reported 0.3 m (1 ft), but the frequency response of the transducer and the control room instrumentation will filter most of the high frequency components, reducing the observed oscillations.

These 0.3 m (1 ft) oscillations will make controlling level in a tight fashion almost impossible when large power oscillations are present and may result in unnecessary low-level isolations in plants without an isolation bypass mechanism in the control room. To avoid this problem, the modified

EPGs require that the water level be reduced to 0.6 m (2 ft) below the spargers; analyses show that following this reduction the large power oscillations are reduced or even eliminated by reduction in core inlet subcooling.

Bypass of MSL and Off-Gas High-Radiation Interlocks

The modified guidelines instruct the operator to bypass the MSL and off-gas high-radiation interlocks to maintain the main condenser as a heat sink, even if there is evidence of gross fuel failure. This bypass is not required to successfully manage any ATWS event when SLC works using either the modified or Rev. 4 EPGs. The main advantage of bypassing this interlock is that the containment heat up rate is reduced (all or part of the heat load is absorbed by the condenser, depending on the bypass valves capacity); thus, ATWS events are likely to have a lower impact on containment.

The BWROG response to a staff Request for Additional Information (RAI) question regarding this issue stated that offsite radiological dose is lessened by using off-gas to scrub the release,[12] but this response was based on a qualitative analysis that did not account for the probability of failures in the balance of plant. The bases for the BWROG response is that containments have a natural leakage rate, so that any radiological release into containment will make its way to the environment. The off-gas system, on the other hand, is designed to scrub quite efficiently radiological releases, including noble gases in some instances; thus, radiological releases to the environment through the off-gas system are minimized.

The reduction in risk for the SLC failure case, however, comes at the cost of increasing the risk for the more common ATWS events when SLC works. Of particular concern for this option is the dependence on a large number of unqualified balance of plant components, and possible conflict among different procedures if leaks are developed in secondary containment.

The advantage of this action from a technical basis may vary from plant-to-plant. Factors that should be considered in any plant- and procedure-specific evaluation of this guideline are:

(1) Comparative off-site radiological consequences with and without MSIV closures.

(2) Impact of balance of plant gross equipment contamination associated with extensive core damage on the accessibility and need for maintenance of critical emergency and supporting service systems.

(3) A plant-specific review to insure that the procedures have positive steps to isolate containment when appropriate.

(4) Capability of timely execution of all bypass functions required by the plant procedures.

Recirculation Pump Runback Prior to Trip

An additional change in the modified EPGs requires the operator to run back the recirculation pumps before they are tripped if the reactor is not isolated. This change in the EPGs should minimize the possibility of level isolations that are possible as a result of the transients induced by tripping the

recirculation pumps, and it should not result in unacceptable delays that may compromise the reactor performance during ATWS because this step only applies if the containment is not isolated.

CONCLUSIONS

The main conclusions from our review are:

(1) Lowering water level below the feedwater spargers to reduce core-inlet subcooling is an effective action to mitigate the consequences of large power oscillations and to reduce suppression pool heat load. For this action to be effective, the water level reduction must be initiated immediately upon ATWS confirmation.

(2) Calculations indicate that higher water levels have a performance advantage over lower levels, and they result in higher margin to HCTL when the SLC system works. Staff calculations show that the advantage for the low-level strategies when SLC fails is small, and both strategies reach the HCTL within approximately 10 minutes.

(3) Calculations indicate that core coolability is not compromised as long the downcomer water level is maintained above the minimum steam cooling water level (MSCWL). Experimental data from the TLTA test confirm these analyses.

(4) Analyses indicate that pressure oscillations induced by either SRV cycling, low downcomer water level, or condensation effects produced by cold feedwater injection are relatively small and do not compromise core coolability or water level measurement instrumentation.

(5) The 1/6 scale facility boron remixing test showed that boron destratification should occur with recirculation flows greater than ~ 10%. Our review of this test has shown that proper scaling laws were used to extrapolate the test results to full scale. Our review of cold-coolant stratification events in operating reactors confirms the general results from the tests and show that stratified boron should remix if the recirculation flow is increased to at least 15%. However, the review of this operational data has revealed an uncertainty with respect to the time constant for destratification, which may be as large as 10 min.

(6) Even allowing for a 10-minute remixing time constant, best estimate staff calculations indicate that all strategies that control level between MSCWL and 2 feet below the feedwater sparger result in a success path.

(7) Reduction of HSBW from the equivalent of 755 ppm (reference to hot) in the Rev. 4 EPGs to 355 ppm (reference to cold, or 487 ppm reference to hot) in the modified EPGs by eliminating unneeded conservatisms is an acceptable way to minimize the possibility of emergency depressurization during ATWS events. The new HSBW results in injection times between 16 and 24 min (depending on performance assumptions), and it is still fairly conservative. In the unlikely case that recriticality occurs after the level is raised, the EPGs require lowering the water level and injection of additional boron solution; therefore, underestimation of boron injection requirements in the HSBW calculations does not result in unacceptable consequences.

(8) During ATWS events, the most critical instrumentation on which the operator relies is the level instrumentation. A series of analyses and tests[11] performed by the Electric Power Research Institute (EPRI) indicate that significant water level oscillations are to be expected if the reactor becomes unstable during an ATWS event; thus, the operator may experience some difficulty controlling the water level tightly, which may result in unnecessary low-level isolations. The modified EPGs require the operator to control water level to 0.6 m (2 feet) below the feedwater spargers, where large power oscillations and the corresponding level oscillations are not expected.[2]

(9) Allowing the recirculation pumps to run back in a controlled fashion before they are tripped manually is an acceptable action if the reactor is not isolated, and it should reduce the probability of unnecessary isolations.

(10) We do not have sufficient data to determine whether bypassing the MSL and off-gas high radiation interlocks is an acceptable action to maintain the main condenser as a heat sink when evidence of gross fuel failure exists. The acceptance of this action from a technical basis would require a plant- and procedure-specific evaluation, which should consider:

(a) Comparative off-site radiological consequences with and without MSIV closures.

(b) Impact of balance of plant gross equipment contamination associated with extensive core damage on the accessibility and need for maintenance of critical emergency and supporting service systems.

(c) A plant-specific review to insure that the procedures have positive steps to isolate containment when appropriate.

(d) Capability of timely execution of all bypass functions required by the plant procedures.

If this evaluation does not support bypassing the interlocks, then manual closure should still be required as needed following a high radiation signal.

ACKNOWLEDGMENTS

All of the TRAC-BF1 and RAMONA calculations reported in this document were performed by Anthony P. Ulses, NRC/NRR/SRXB.

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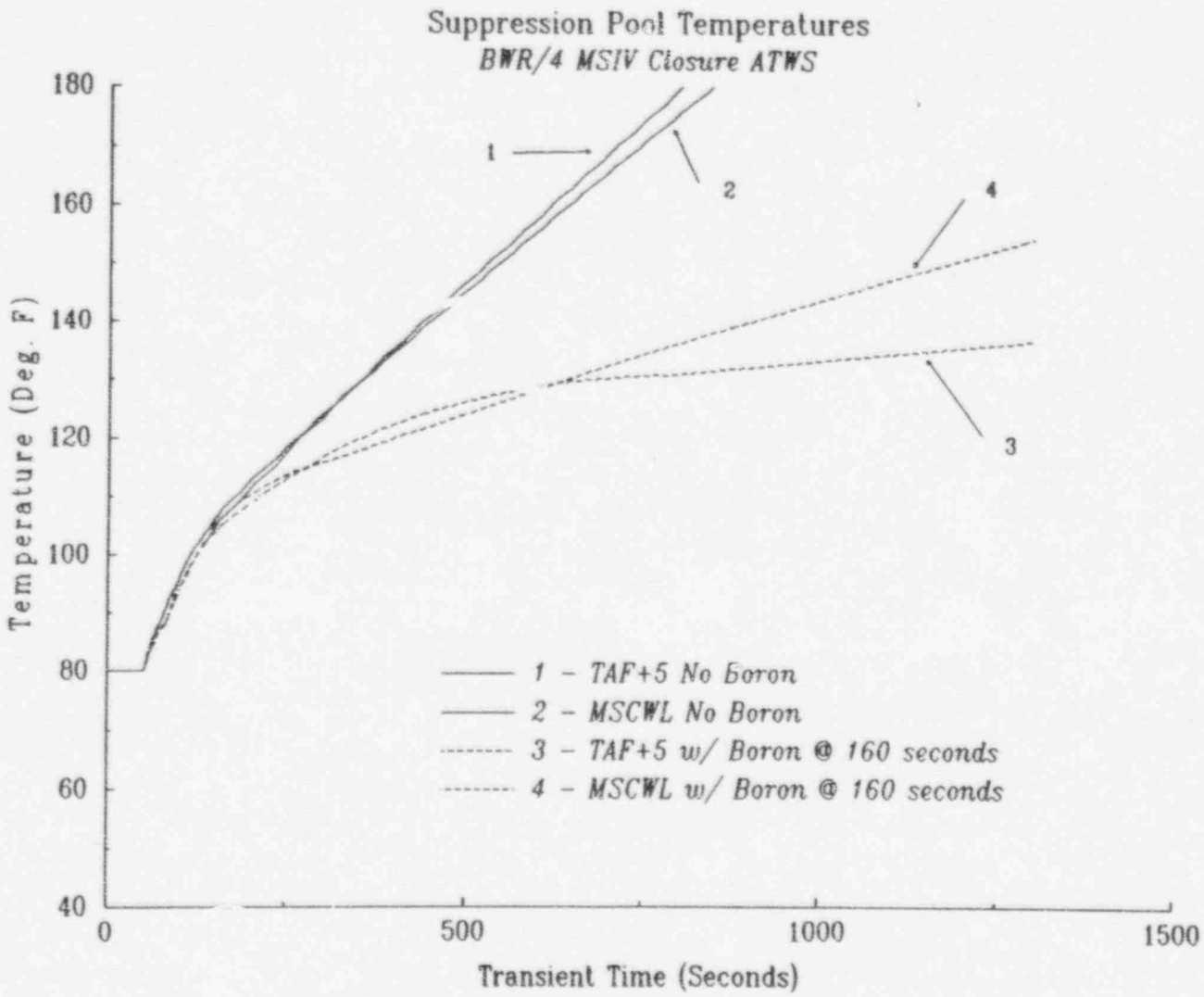

Figure 1. Suppression pool temperatures calculated by TRAC-BF1 following a MSIV isolation ATWS.Downcomer level is maintained at either MSCWL or TAF + 5'.

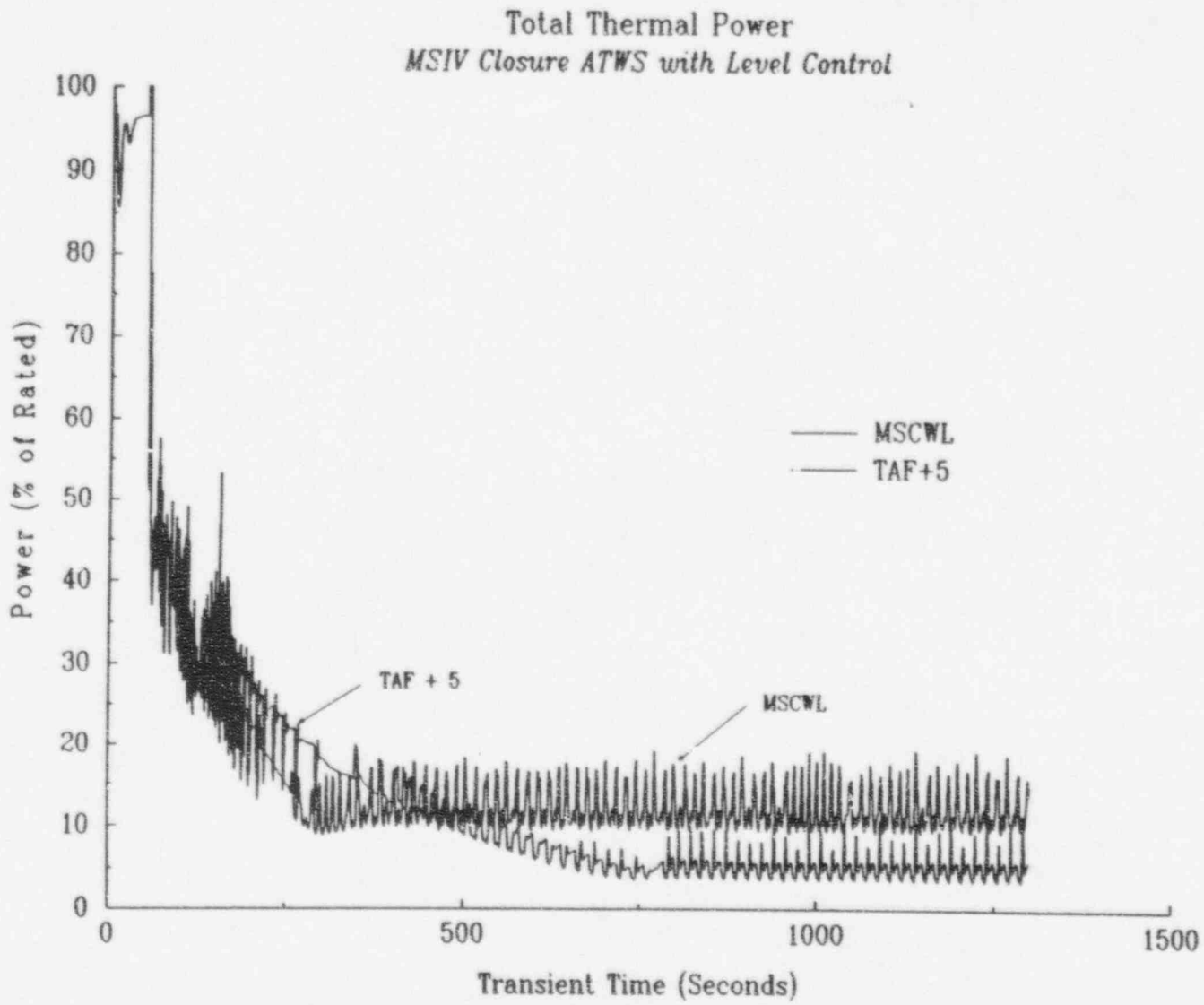

Figure 2. Comparison of power levels during a MSIV closure ATWS with water level control at MSCWL versus TAF + 5'. Power oscillations are caused by SRV actuation.

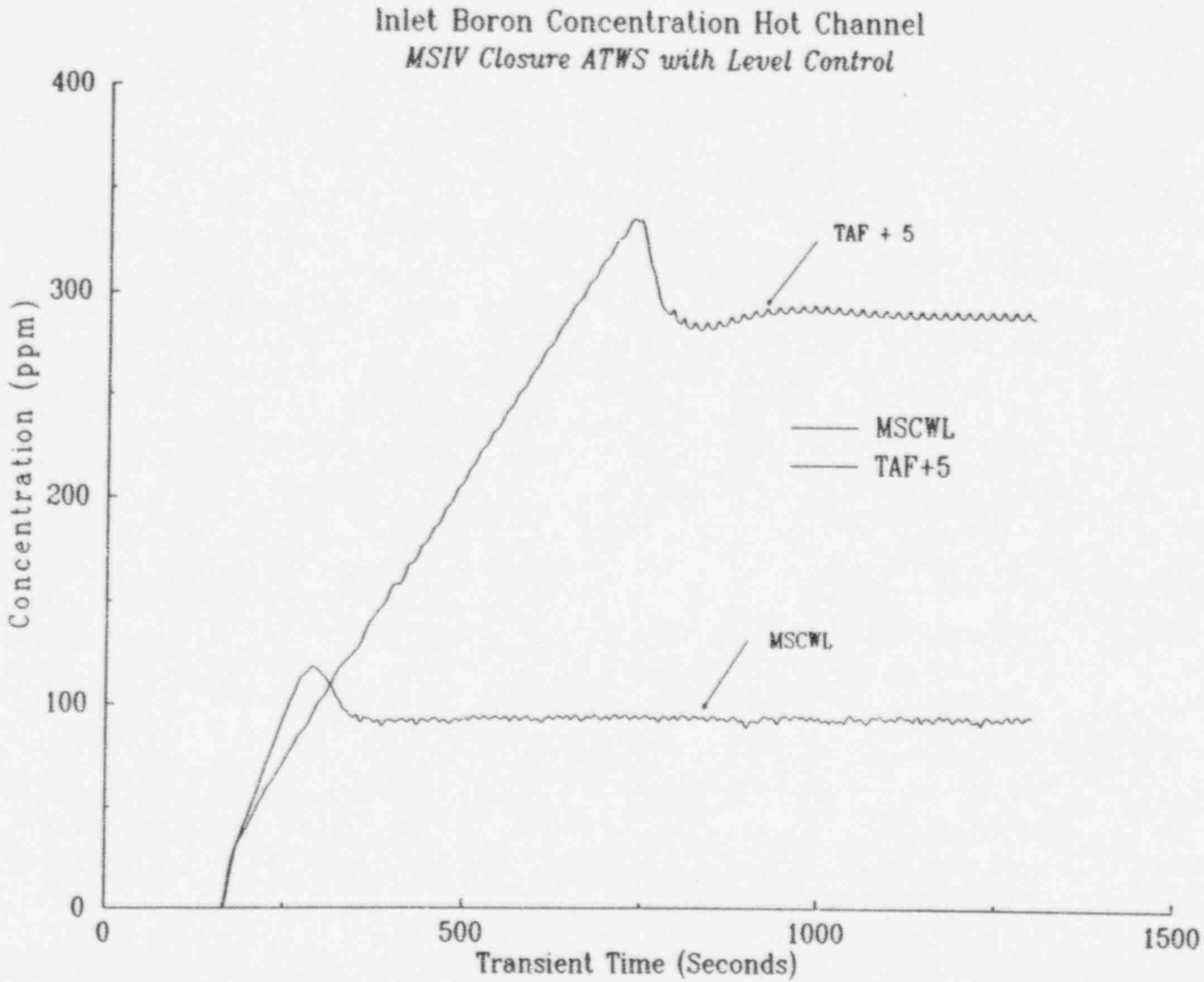

Figure 3. Comparison of boron concentration for a MSIV closure ATWS event with two different level control strategies.

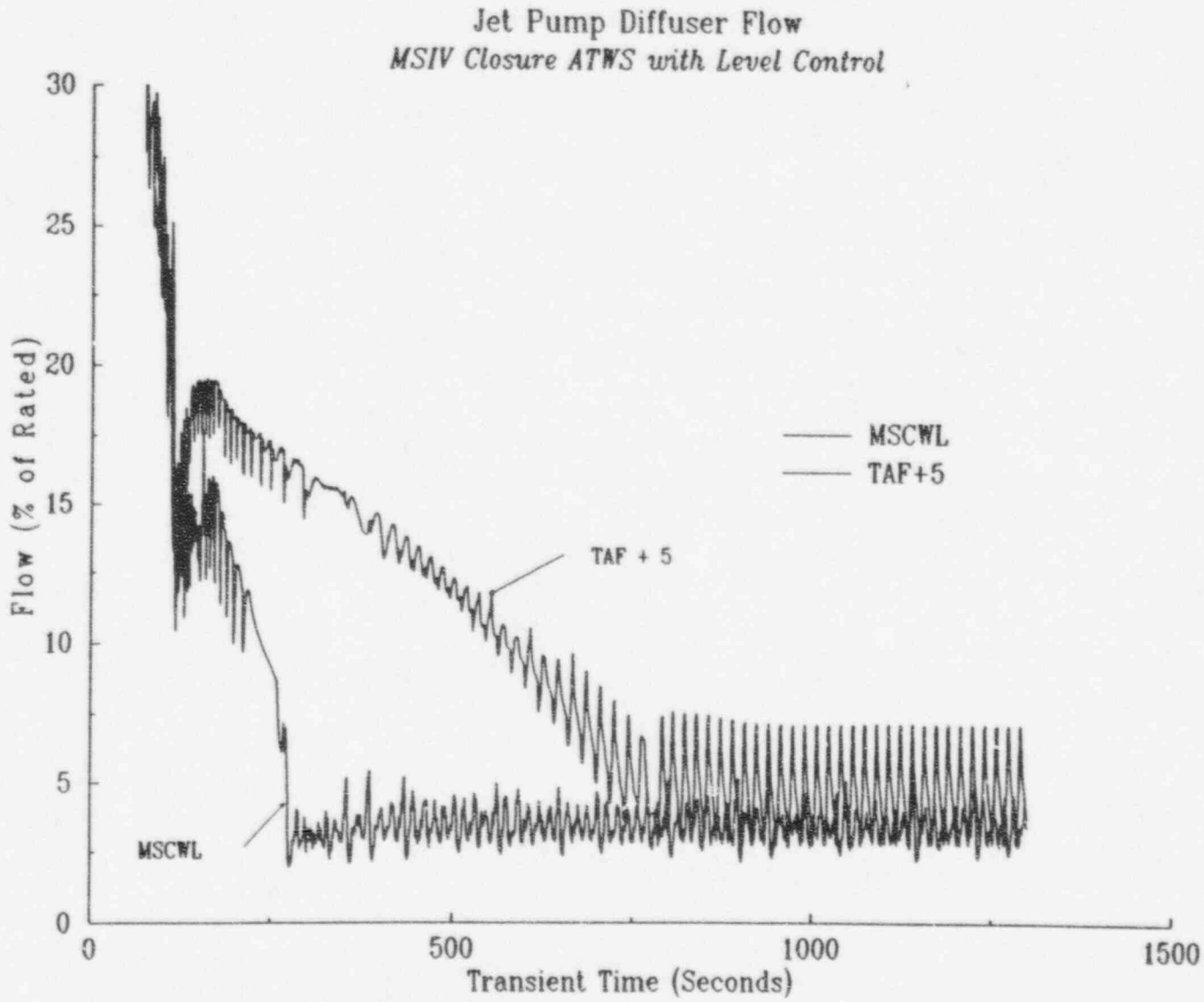

Figure 4. Comparison of core inlet flow for a MSIV closure ATWS event with two different level control strategies.

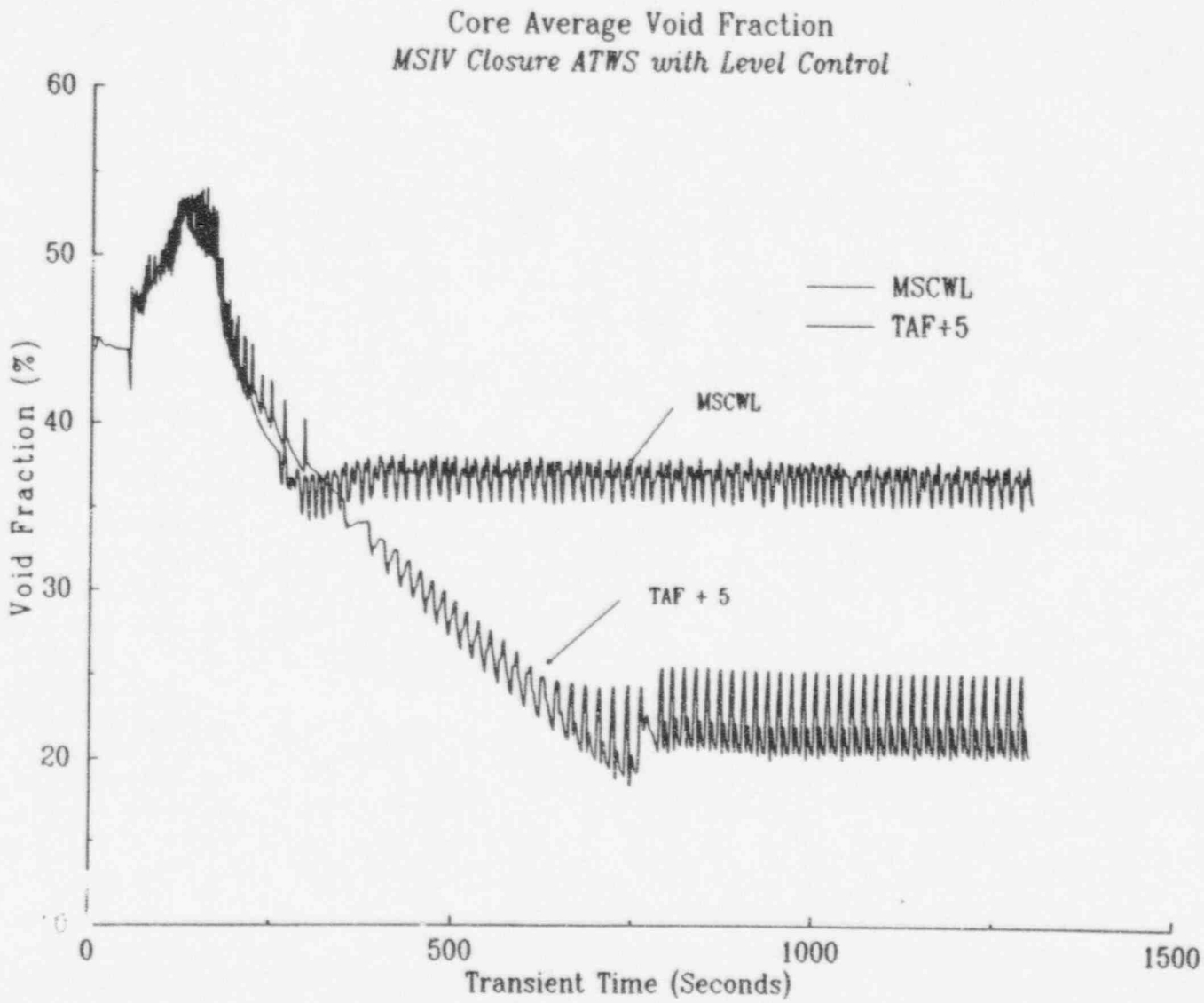

Figure 5. Comparison of core average void fraction for a MSIV closure ATWS event with two different level control strategies.

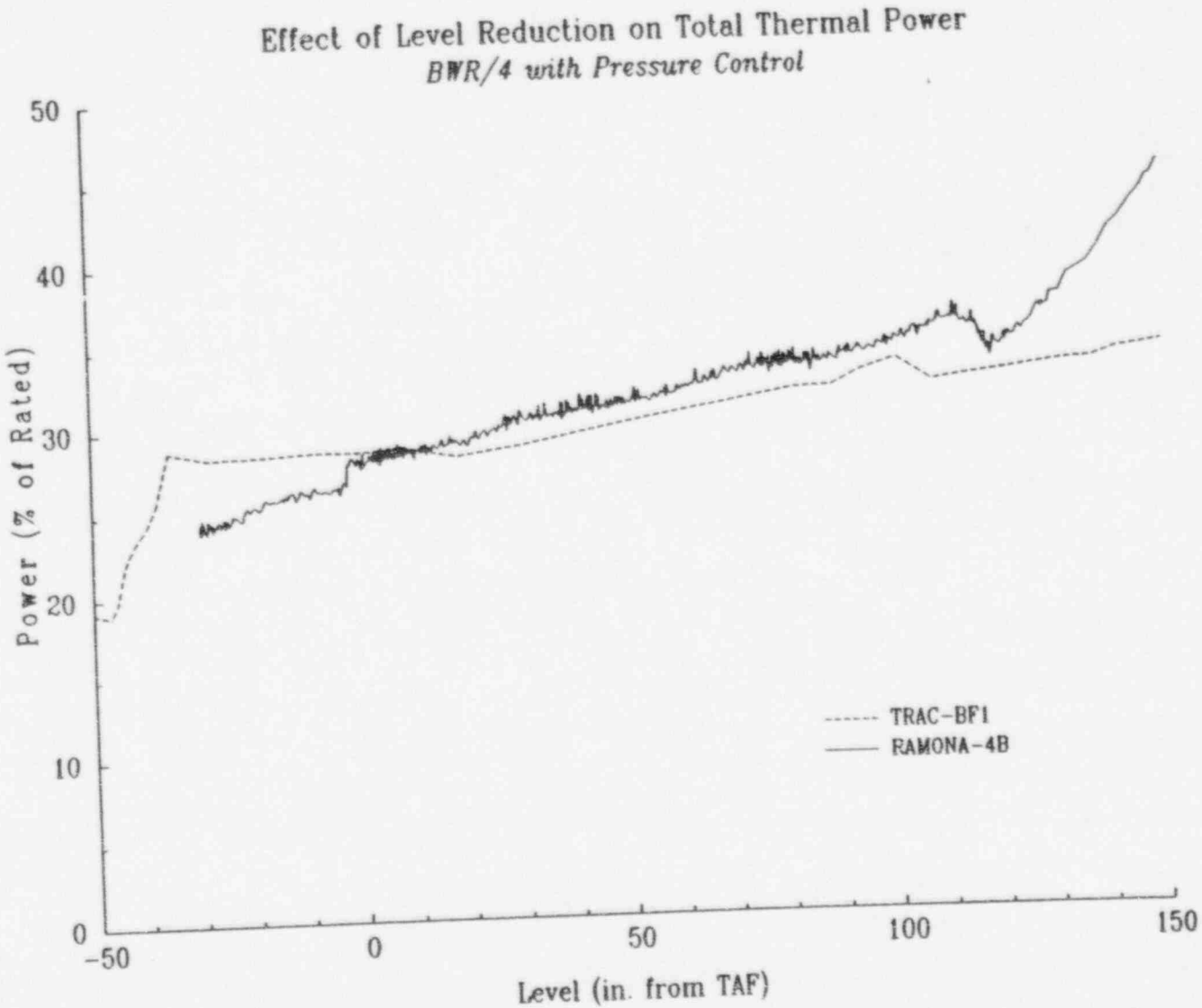

Figure 6. TRAC-BF1 and RAMONA-4B calculated power as a function of downcomer water level with 25 °C feedwater. Sharp drop in power at TAF - 40" (TRAC-BF1 calculation) corresponds to flow stagnation.

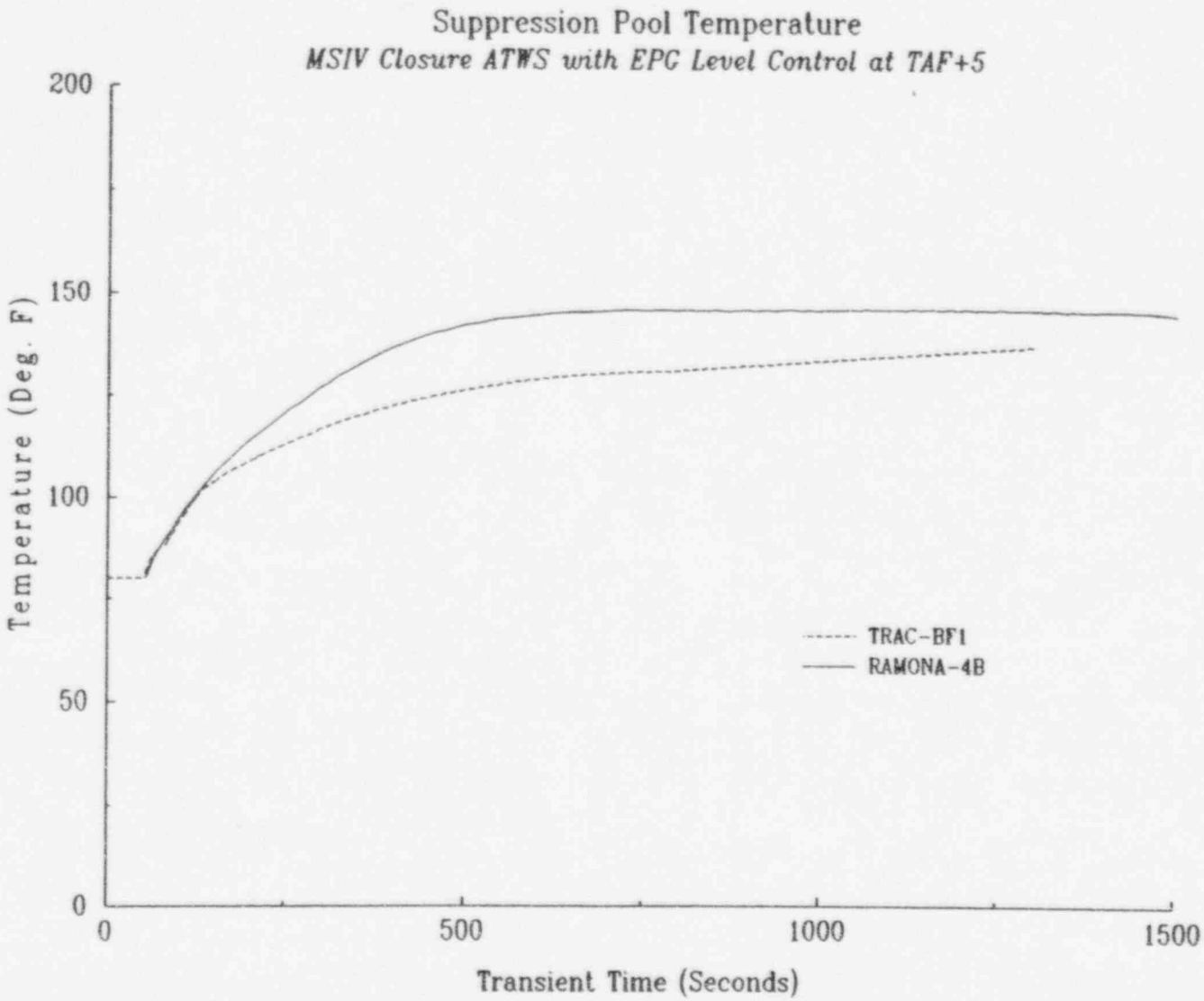

Figure 7. TRAC-BF1 and RAMONA-4B calculated suppression pool temperatures using EPG level control with boron injection.

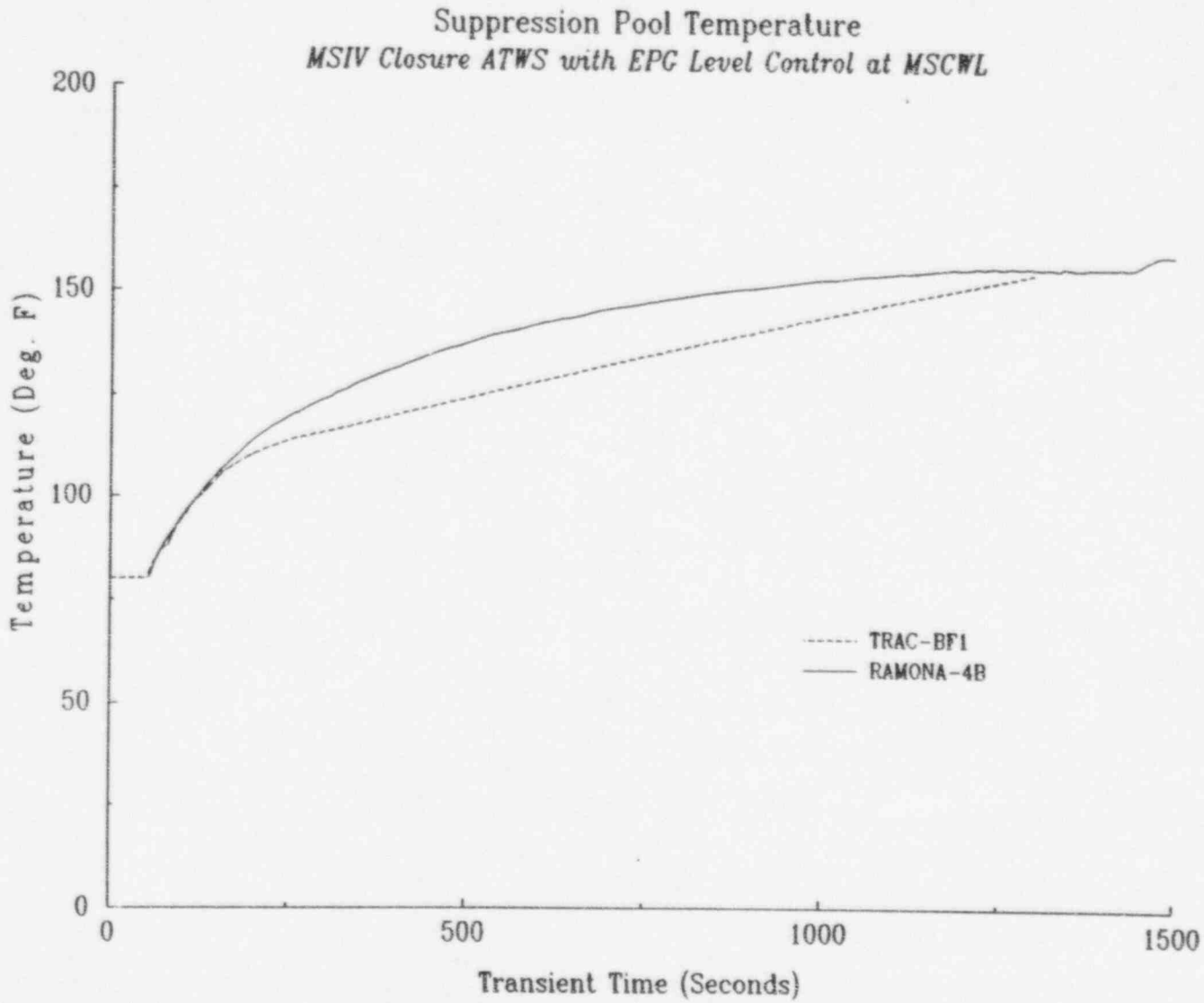

Figure 8. TRAC-BF1 and RAMONA-4B calculated suppression pool temperatures using EPG level control with boron injection.

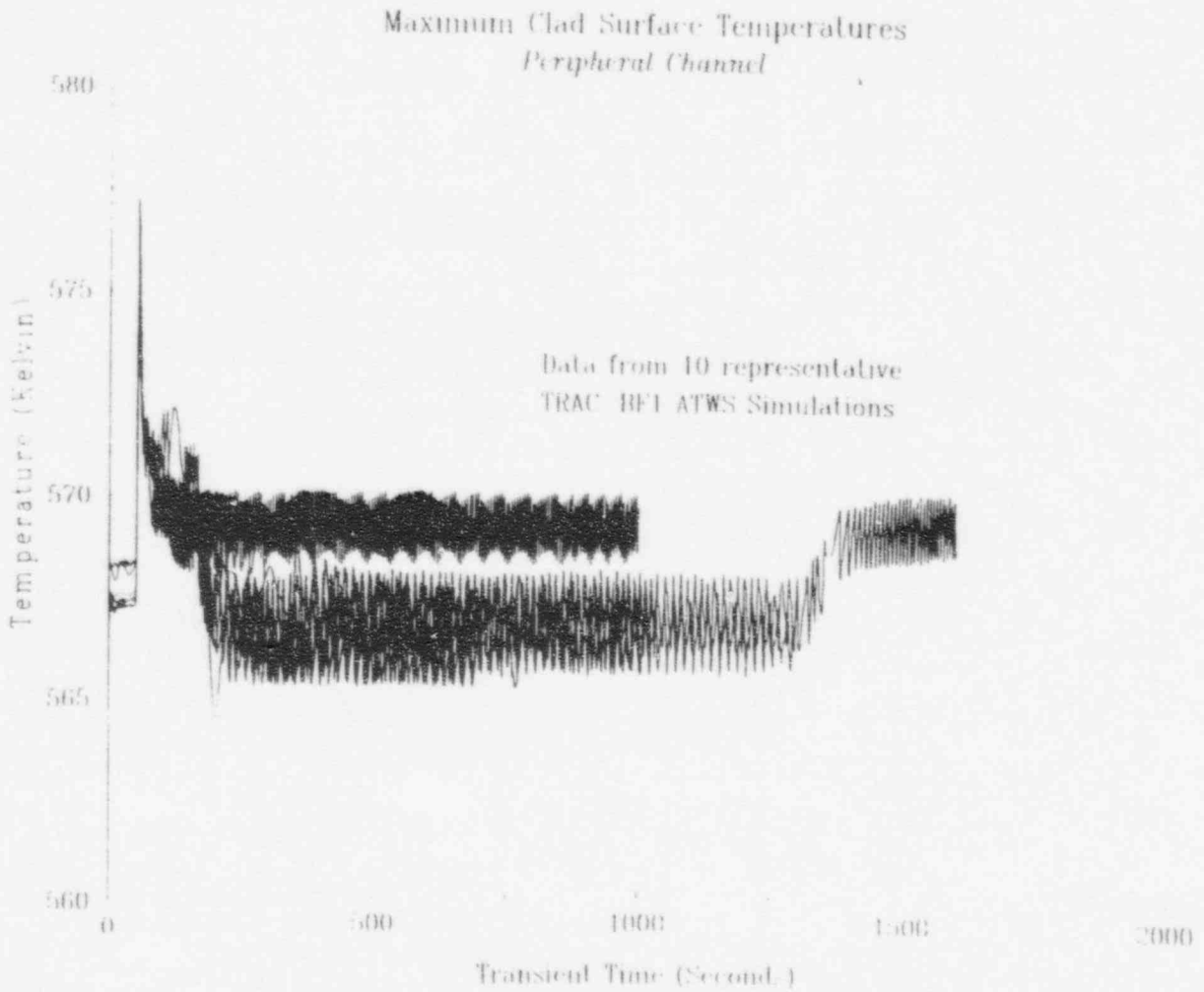

Figure 9. Composite of ten different TRAC-BF1 ATWS calculations showing the maximum clad temperature of the low-power channel.

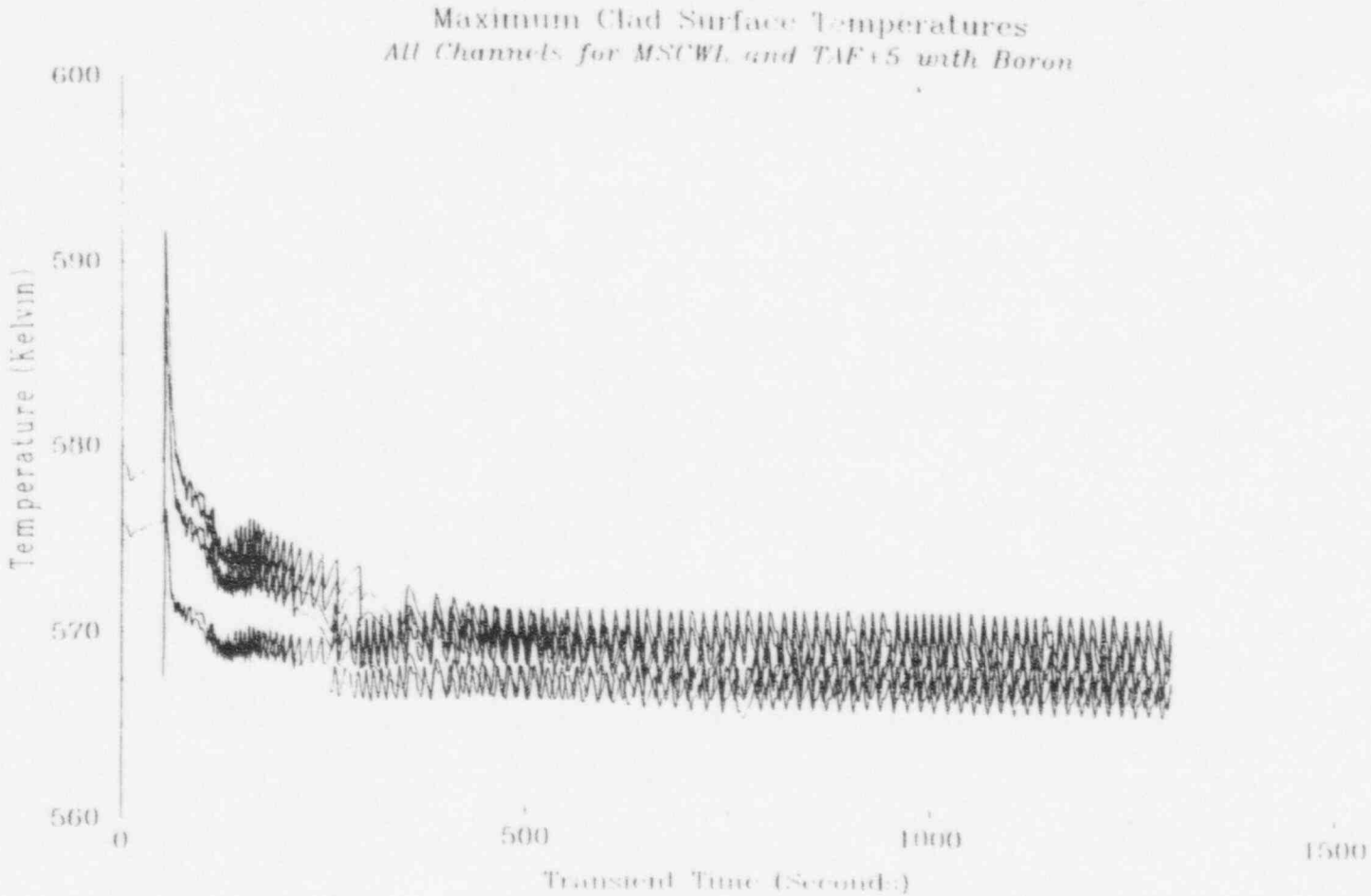

Figure 10. Peak clad temperature for all analyzed channels. TRAC-BF1 for two level control strategies: MSCWL and TAF + 5' with boron.

Figure 11. Boron mixing correlation used for old BWORG analyses and new data from ref. 3

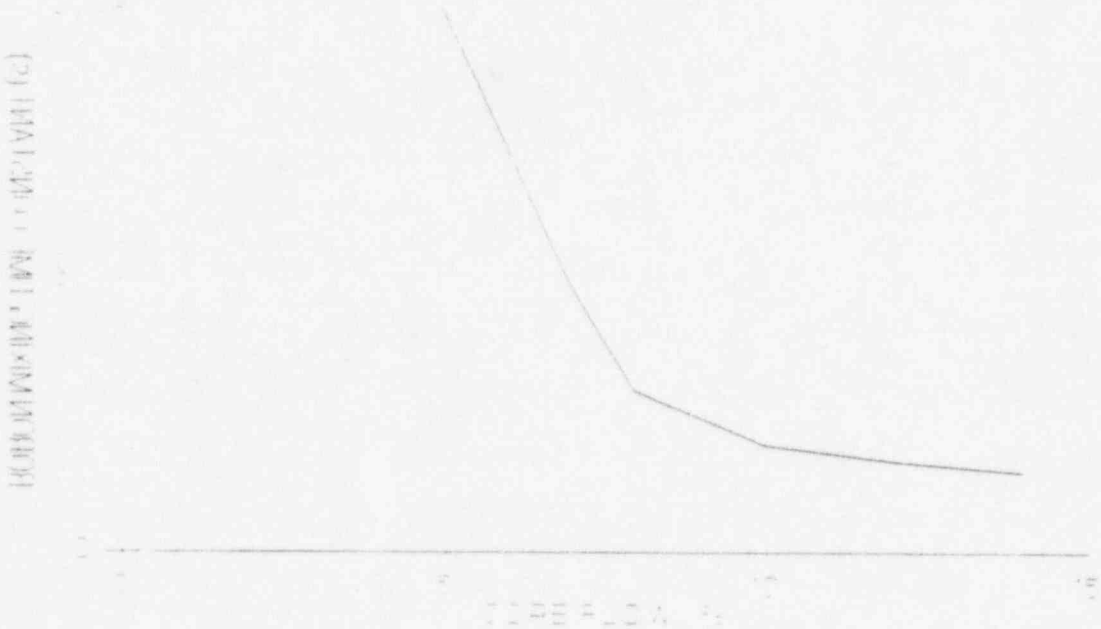

Figure 12. Bor mixing time constant correlation from ref. 6

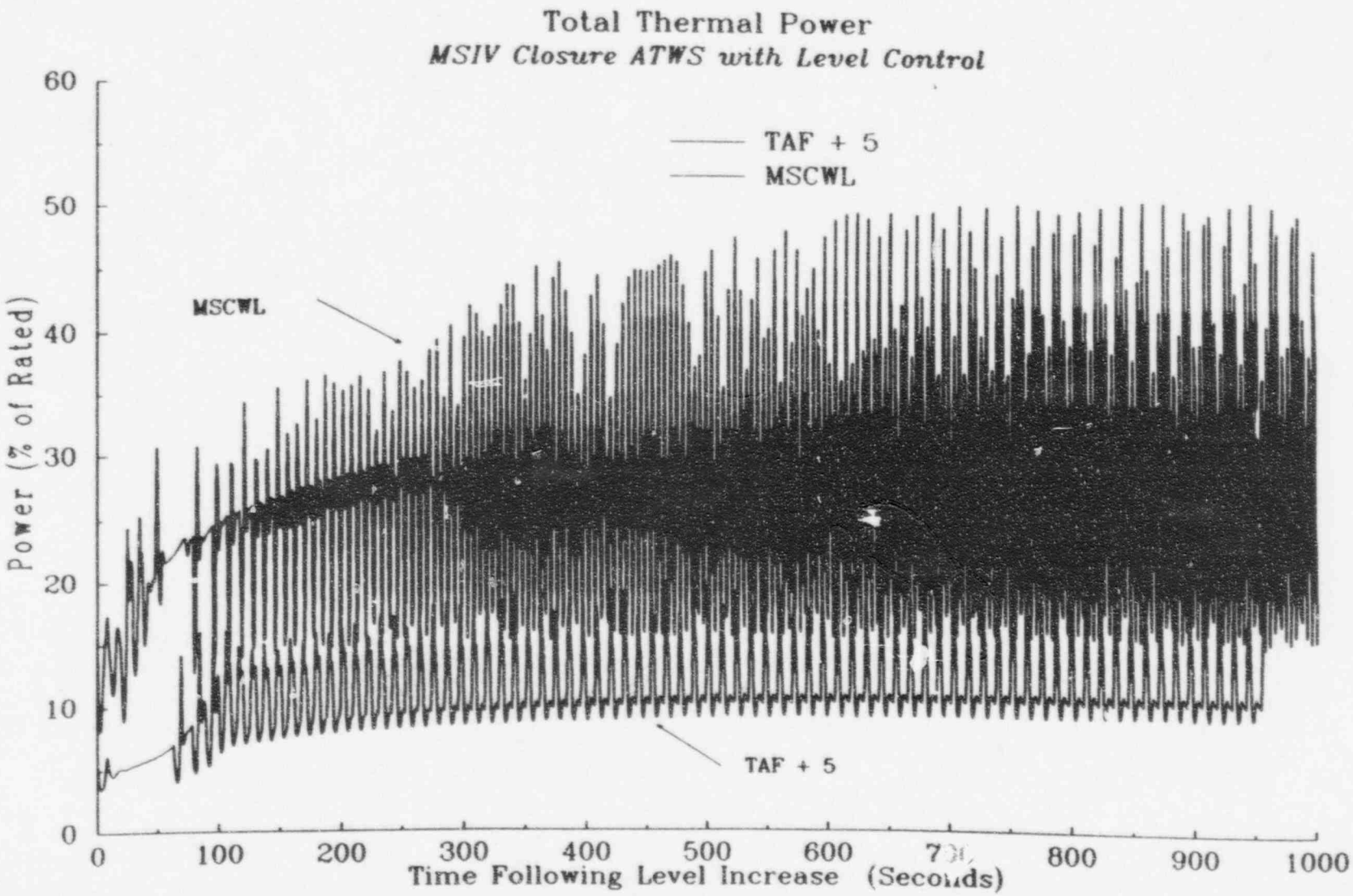

Figure 13. Core thermal power following level increase assuming that boron does not remix.

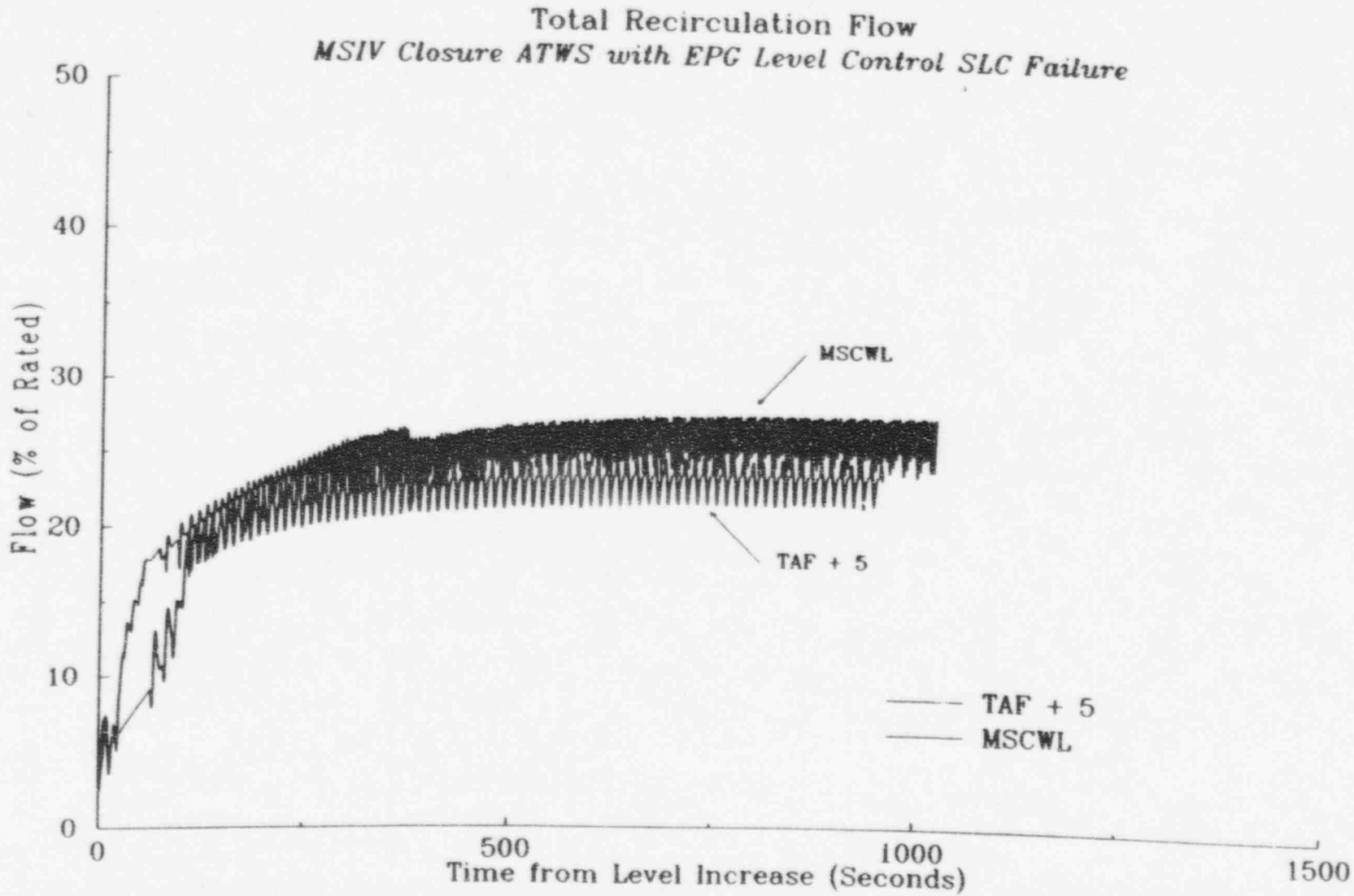

Figure 14. Recirculation flow following level increase assuming that boron does not remix.

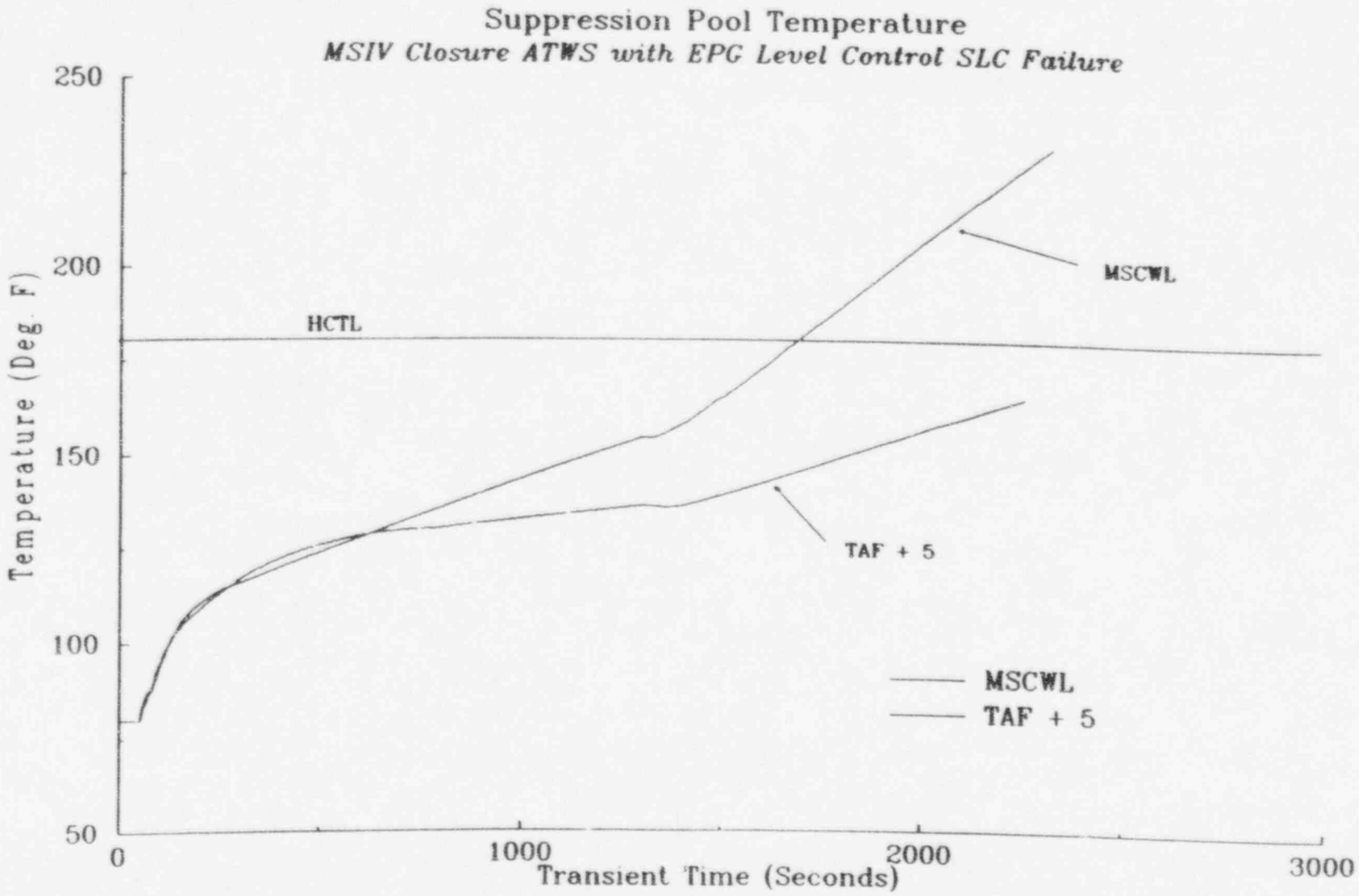

Figure 15. Suppression pool temperature assuming level increase without boron remixing at 1300 seconds.

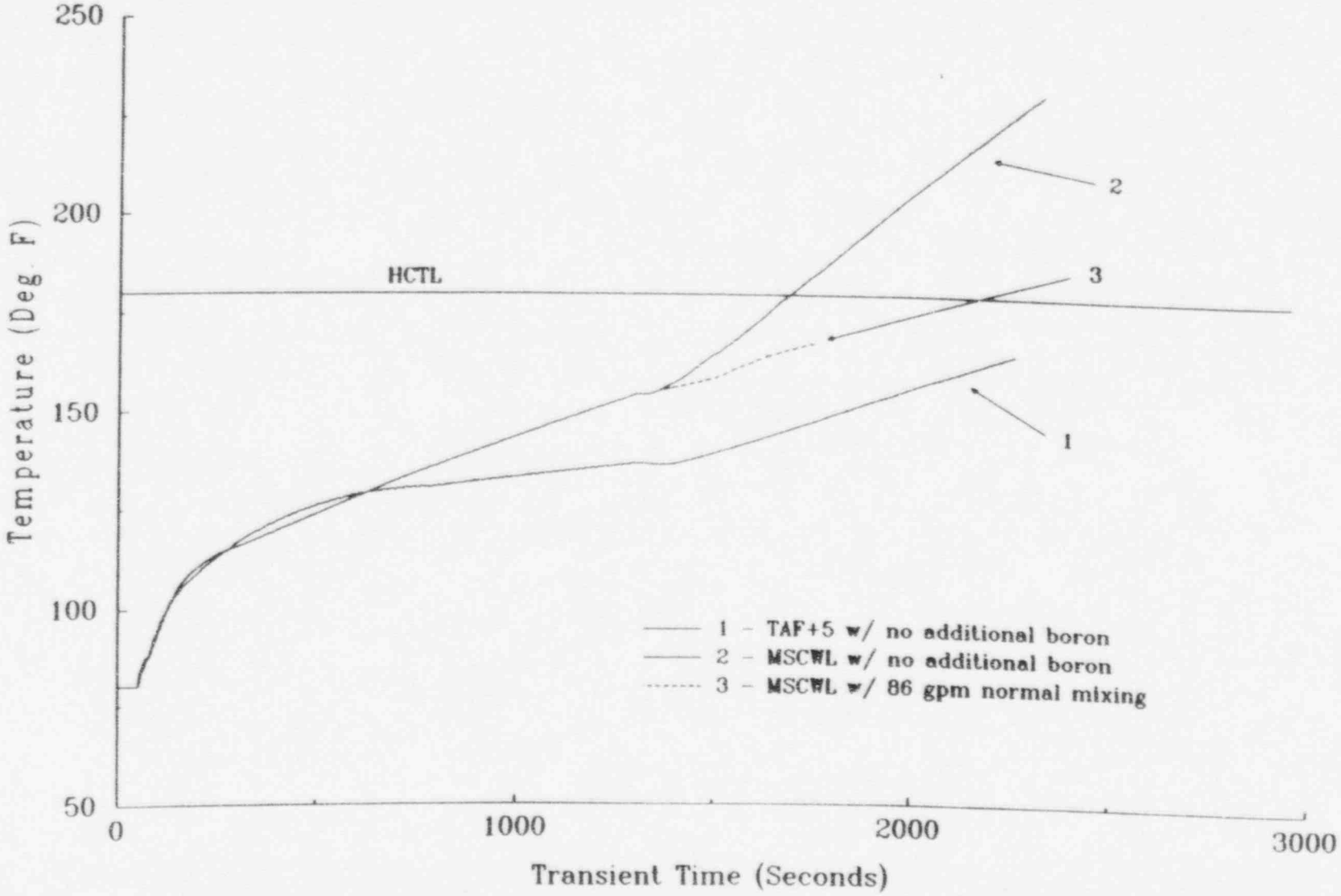

Figure 16. Suppression pool temperature assuming level increase without boron remixing, but with additional SLC injection (86 gpm).

APPENDIX A

AUDIT OF OEI's CODE - CRAC

On July 27, 1994, an audit team composed of two NRC staff members, Laurence E. Phillips and Anthony P. Ulses, and a NRC contractor, Jose A. March-Leuba, performed an audit of the Containment Response Analysis Code (CRAC). This code has been developed by Operations Engineering Inc. (OEI), and it has been used as the analytical basis to evaluate the modified ATWS EPG's. Present at the audit were four OEI representatives, Taggart Rogers, Jim Schilder, Steve Oswald, and Ken Ross.

Documents reviewed included:

- OEI Document 9402-3. "The Management of ATWS by Boron Injection and Water Level Control"
- OEI Document 9402-2. "Containment Response Analysis Code Functional Description"
- OEI Document 9402-6. "Containment Response Analysis Code User's Manual"

CRAC DESCRIPTION

CRAC's main objective is to estimate the containment response; for this reason, the core models are very simplified. CRAC calculations are performed in a spreadsheet and assume quasi-equilibrium conditions at every time step. The calculation procedure for each time step can be summarized as follows:

- The core average void fraction is determined based on a reactivity balance, which uses the initial average void fraction (defined by the initial control rod line), the void reactivity coefficient and the boron reactivity (proportional to the boron concentration).
- The core inlet mass flow rate is estimated to be proportional to the square root of the pressure drop differential between the shroud and the downcomer. The constant of proportionality is defined as function of downcomer water level, and it is calculated by CRAC to match the power-flow-level correlation table that is input to the code by the user.
- The input mass flow rate and the void fraction define a core-average liquid velocity. The steam velocity is estimated by adding (as opposed to ratioing) a slip velocity, which CRAC assumes to be a function of downcomer water level (as opposed to a function of the void fraction). CRAC computes this slip velocity to match the user input power-flow-level correlation table at the input levels.
- The core steam flow is then calculated by CRAC as the product of the steam velocity times the flow area times the core-average void fraction.
- The reactor pressure vessel (RPV) steam flow is significantly smaller than the core steam flow if cold feedwater is being injected in the steam area, causing steam condensation.

- If MSIV's are closed, all the RPV steam flow is dumped into the suppression pool, and the pool temperature is computed from an energy balance assuming perfect condensation.

- CRAC assumes that the containment pressure is in equilibrium with the temperature of the suppression pool. For example, CRAC assumes that hard containment venting initiates when the suppression pool reaches 295°F, the saturation temperature that corresponds to 60 psig.

IDENTIFIED DEFICIENCIES

Two main deficiencies were identified in CRAC by the audit team: (1) the slip model, and (2) the numbers used as default power-flow-level correlation for the 100% rod line case. In the judgement of the audit team, these two deficiencies make CRAC results overly conservative, especially for the case of "Strategy A".

Slip Model

The slip model in CRAC may result in significant inaccuracies. CRAC uses a core-average void and a delta slip (as opposed to a ratio); therefore, the computed steam velocities are very high even if the core flow were negligible. Table A.1 shows some of the numbers computed by CRAC for the 100% rod line case reported in ref. 4.

We see from Table A.1 that, given CRAC's calculation procedure, even if the core flow was significantly reduced (e.g., by 50%), the steam flow would be essentially unchanged because the additive slip component (13.87 ft/s) is ~10 times larger than the flow-dependent component (1.87 ft/s). The only way that CRAC can reduce outlet steam flow is by reducing the average void fraction. Reductions in flow do not affect the outlet steam flow.

This slip model unfairly penalizes Strategy A because it forces it to have twice as much steam flow than strategy B (18.87 ft/s slip velocity for strategy A versus 8.56 ft/s slip for B), even if the core inlet flow and core average void fraction were the same for both strategies. This appears to be the root cause for the large difference in performance observed by CRAC between the two strategies. When strategy A is compared to B in ref. 4, the core power "stagnates" at ~25% for strategy A (Fig. 5.1.2-6 of ref. 4), but only at 17% for strategy B (Fig. 5.1.1-6 of ref. 4). This difference in power is hard to justify on first principles when one considers that both strategies have stagnated recirculation flow and strategy A has 200 ppm boron concentration and a lower void fraction than strategy B (100 ppm boron).

Independent audit calculations performed by NRC indicate that the trend and the differences between the two strategies are exactly opposite to the ones calculated by CRAC. As shown in Figs. B.1 and B.2, these NRC scoping analyses result in a lower stagnation power for strategy A (level at TAF + 60") than for strategy B (Level at TAF), and a lower final suppression pool temperature. Table A.2 shows a comparison between the results of CRAC and NRC scoping calculations.

Table A.1 Steam and Slip Velocities for the 100% rod line case of ref. 11

Water level =	TAF (Strategy B)	TAF + 60" (Strategy A)	Nominal
Upper Plenum Void (Input by user)	77%	77%	77%
Core Inlet Flow (Input by user)	9%	18%	30%
RPV Steam Flow (Input by user)	15%	25%	45%
Condensed Steam Flow	10.1%	16.8%	0%
Core Exit Steam Flow	25.1%	41.8%	45%
Core Liquid velocity	0.86 ft/s	1.87 ft/s	3.6 ft/s
Estimated Slip Velocity	8.56 ft/s	13.87 ft/s	13.32 ft/s
Core steam velocity	9.44 ft/s	15.74 ft/s	16.93 ft/s

Table A.2. Comparison between CRAC and preliminary NRC audit results for MSIV-isolation ATWS from the 100% rod line

	STRATEGY A (TAF + 60")		STRATEGY B (TAF)	
	CRAC[11]	NRC audit	CRAC[11]	NRC audit
Core stagnation power	~25%	11%	~17%	12%
RPV stagnation steam flow	~15%	8.4%	~10%	10%
Max. pool temperature	162°F	144°F	148°F	150°F
Initial pool temperature	80°F	80°F	80°F	80°F

CRAC Power-Flow-Level Correlation

As seen in Table A.2, CRAC calculates significantly higher stagnation powers than those estimated by preliminary NRC audit calculations. The CRAC stagnation powers are also significantly larger than those assumed in previous ATWS analyses, for which the stagnation power was assumed "conservatively" to be 8%. From our review, we conclude that these high powers are mostly due to the slip model discussed above and to the highly conservative power-flow-level correlation used.

The power-flow-level correlation is input by the user in CRAC. The correlation used for the ref. 4 analyses is documented in Table 3.1-1 of ref. 4. These values are also reproduced in Table A.1 of

this report. We see that the input correlation "tells" CRAC that the RPV steam flow is 25% at TAF + 60" (the strategy A level). CRAC then estimates that the condensed steam is 16.8%, so that the core power should be 41.8%. The problem is that the core power at the nominal level (TAF + 161") is given by the correlation as 45%. Therefore, the correlation used by CRAC in ref. 4 basically says that a water level reduction of 100", with the very significant associated reduction of subcooling only accounts for a power reduction of 3.2%. Then, a further level reduction of 60" to TAF with no reduction of subcooling accounts for a reduction of 16.7% in power (from 41.8% to 25.1%). This situation is extremely conservative and penalizes the strategy A power by giving it essentially the same power as under nominal water level.

APPENDIX B

PLANT CLASSIFICATION

There are differences in emergency equipment between different types of BWRs, which makes the management and possible consequences of ATWS events different among these plants. Of special relevance to ATWS events is the SLC injection point (inside or outside the shroud), and the availability of a high-pressure injection (HPCI) in the emergency core cooling system (ECCS).

Classification Based on SLC Injection Point

The largest difficulty with ATWS management occurs in plants with SLC injection outside the shroud because of the sodium pentaborate stratification caused by flow stagnation. In the plants with SLC injection inside the shroud, the sodium pentaborate solution is injected by the core spray on top of the core, so when the flow stagnates, it drops inside the core where it is mixed with the turbulent boiling flow. In these plants, the boron concentration in the core reaches shutdown levels well before the HSBW is injected, and ATWS with SLC are of little or no consequence if the water level is dropped sufficiently to stagnate flow. Boron injection inside the shroud occurs in all BWR-6's and in all BWR-5's except one site (two units). Table B.1 shows a summary of characteristics of BWR-5's and BWR-6's.

Table B.1. ATWS-related parameters for BWR-5's and BWR-6's

Plant	Boron inside the shroud ?	Electric-driven FW pumps?	Key-lock isolation bypass?	Distance FW sparger to MSIV isolation[a]
A	No	Yes	No	95"
B	Yes	Yes	No	94"
C	Yes	No	Planned	16"
D	Yes	Yes	No	110"
E	Yes	No	No	111"
F	Yes	Yes	Yes	108"
G	Yes	Yes	Yes	106"

[a] Requires a minimum of 24" (36" including instrument uncertainty) to avoid isolation under new EPG procedures

Classification Based on Availability of Controllable High Pressure Injection

A significant problem in plants without HPCI is that flow out of the high pressure core spray (HPCS) system cannot be controlled (it is either on or off), and the EPGs specifically instruct the operator not to use HPCS for level control. The other high-pressure emergency coolant sources cannot provide sufficient coolant to maintain water level above the MSCWL, even at the flow-stagnation power. The only other high-pressure source of coolant that can be used to control level in this type of plants is feedwater, but steam supply to drive the feedwater pumps cannot be guaranteed if the turbine trips.

As shown in Table B.1, of the seven plants (all BWR-5's and BWR-6's) without HPCI, five have electric-motor-driven feedwater pumps that do not require turbine steam. These five plants have sufficient high-pressure coolant supply to control water level at or above TAF. The two plants with 100% steam-driven feedwater pumps inject boron inside the shroud, so that the reactor shuts down early into the transient and water level does not drop below MSCWL.

Based on some ATWS analyses performed at the BWR-6 simulator in the NRC Technical Training Center, we concluded in a previous report[8] that plants without HPCI may have problems following either the Rev. 4 or the modified EPGs, because they could not provide sufficient high pressure injection to hold level. Recent analyses performed by the BWROG appear to indicate that boron injection inside the shroud shuts down the reactor so fast that emergency depressurization is not required. The non-HPCI plant that injects boron through the stand pipes and do not shut down early has sufficient electric-motor driven feedwater pump capacity to hold level. Thus, these plant-specific characteristics resolve the issue, and we concur with the BWROG position that the seven non-HPCI plants are likely to avoid emergency depressurization during a standard ATWS with SLC functional.

Classification Based on Margin to MSIV Isolation Water Level

The main reservation with the actions proposed in the modified EPGs is that, by lowering the water level, there is an increased probability that the reactor may be isolated because of a low water level signal. Most plants have either sufficient margin to the MSIV isolation level or have a switch and a procedure to bypass the isolation function in the control room. It is expected that plants without bypass in the control room may take as long as 30 minutes to hardwire a bypass in the plant; thus non-control-room isolation bypasses are of no significant help during most ATWS events.

The BWROG has submitted a list of plants with their margin to MSIV isolation level and the availability of bypass in the control room. We enclose this list in this report as Table B.2. In view of recent water-level instrumentation studies,[11] we should expect water level oscillations on the order of 0.3 m (1 ft); thus, plants C, E, and H in Table B.2 are likely to suffer full MSL isolations when following the modified EPGs. These three plants do not have a control room isolation bypass; therefore, these three plants will require outstanding operator performance to avoid MSL isolations if they follow the modified EPGs. It is expected that installing a key-lock bypass will improve ATWS response significantly, and it will result in reduced risk for these plants.

Plants A, B, and U in Table B.2 have the MSIV isolation level higher than 0.6 m (24 inches) below the spargers, but these plants have (or have planned) control room isolation bypass, so isolation is not likely.

Table B.2. Margin to isolation trip level when controlling level 0.6 m (2 ft) below the feedwater spargers

Plant	Isolation Bypass	Top of new control band to MSIV isolation (inches)	Comments
A	Y	-46	
B	Y	-45	
C	N	12	Not enough margin
D	Y	10	
E	N	11	Not enough margin
F	N	18	
G	N	18	
H	N	3	Not enough margin
I	Y	29	
J	N	37	
K	N	48	
L	N	44	
M	N	44	
N	N	74	
O	N	97	
P	N	81	
Q	N	76	
R	N	71	
S	N	76	
T	N	71	
U	Planned	-9	
V	N	70	
W	Y	82	
X	N	86	
Y	Y	84	
Z	N	87	