PWROG
PWR Owners Group

Domestic Members
AmerenUE
Callaway
American Electric Power Co.
D.C. Cook 1 & 2
Arizona Public Service Co.
Palo Verde 1, 2 & 3
Constellation Energy Group
Calvert Cliffs 1 & 2
R. E. Ginna
Dominion Kewaunee
Dominion Nuclear Connecticut
Millstone 2 & 3
Dominion Virginia Power
North Anna 1 & 2
Surry 1 & 2
Duke Energy
Catawba 1 & 2
McGuire 1 & 2
Oconee 1, 2, 3
Entergy Nuclear Northeast
Indian Point 2 & 3
Entergy Nuclear South
ANO 1
ANO 2
Waterford 3
Exelon Generation Company LLC
Braidwood 1 & 2
Byron 1 & 2
Three Mile Island 1
FirstEnergy Nuclear Operating Co.
Beaver Valley 1 & 2
Davis Besse
FPL Group
St. Lucie 1 & 2
Seabrook
Turkey Point 3 & 4
Nuclear Management Co.
Palisades
Point Beach 1 & 2
Prairie Island 1 & 2
Omaha Public Power District
Fort Calhoun
Pacific Gas & Electric Co.
Diablo Canyon 1 & 2
Progress Energy
Crystal River 3
H. B. Robinson 2
Shearon Harris
PSEG – Nuclear
Salem 1 & 2
South Carolina Electric & Gas Co.
V. C. Summer
Southern California Edison
SONGS 2 & 3
STP Nuclear Operating Co.
South Texas Project 1 & 2
Southern Nuclear Operating Co.
J. M. Farley 1 & 2
A. W. Vogtle 1 & 2
Tennessee Valley Authority
Sequoyah 1 & 2
Watts Bar 1
TXU Power
Comanche Peak 1 & 2
Wolf Creek Nuclear Operating Corp.
Wolf Creek

International Members
British Energy plc
Sizewell B
Electrabel
Doel 1, 2, 4
Tihange 1 & 3
Electricité de France
Kansai Electric Power Co.
Mihama 1
Takahama 1
Ohi 1 & 2
Korea Hydro & Nuclear Power Co.
Kori 1 – 4
Ulchin 3 - 6
Yonggwang 1 - 6
NEK
Krško
NOK
Kernkraftwerk Beznau
Ringhals AB
Ringhals 2 – 4
Spanish Utilities
Asco 1 & 2
Vandellos 2
Almaraz 1 & 2
Taiwan Power Co.
Maanshan 1 & 2

Project No. 694
CENPD-254-P

June 19, 2006

OG-06-200

Mr. Daniel S. Collins
U. S. Nuclear Regulatory Commission
Special Projects Branch
Office of Nuclear Reactor Regulation
One White Flint North
11555 Rockville
Rockville, MD 20852-2738

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: **Suspension of NRC Approval for Use of Westinghouse Topical Report CENPD-254-P, Post LOCA Long Term Cooling Model, Due to Discovery of Non-Conservative Modeling Assumptions During Calculation Audit, PA-ASC-0290**

References:

1.) NRC letter dated November 23, 2005, D.S. Collins to G. C. Bischoff, "Suspension of NRC Approval for Use of Westinghouse Topical Report CENPD-254-P, Post LOCA Long Term Cooling Model, Due to Discovery of Non-Conservative Modeling Assumptions During Calculation Audit".

2.) Conference call with the Westinghouse Owners Group and NRC on December 7, 2005.

Dear Mr. Collins:

The purpose of this letter is to respond to the Reference 1 letter requesting written response to concerns associated with boric acid analyses following a loss-of-coolant accident (LOCA). Specifically, Reference 1 requested that licensees who have relied on CENPD-254-P or similar analytical models perform an evaluation to confirm that sufficient margin exists and that they remain in compliance with the regulations and their design basis. A follow-up phone conference with the NRC (Reference 2) clarified NRC expectations regarding this request.

D048

In response to the NRC request, and in accordance with the stated NRC expectations, the PWR Owners Group funded a program to review and qualitatively evaluate the boric acid precipitation analysis-of-record (AOR) for each participating licensee to confirm that post-LOCA boric acid precipitation would not occur and that regulatory requirements are met. As suggested by the staff, these evaluations used insights from analyses performed for the Waterford 3 Extended Power Uprate, along with recent analyses for the Beaver Valley and Ginna Extended Power Uprates, to show that existing calculations are conservative and post-LOCA boric acid precipitation would not occur. Additionally, the program included a safety assessment of all post LOCA boric acid precipitation issues identified by the NRC. Since, the NRC concerns cited in Reference 1 apply to all three US PWR designs (W, CE and B&W), this program required the participation of both US PWR safety analysis vendors (Westinghouse and AREVA NP) as well several licensees who perform their own analyses. The list of licensees who are represented in this response is provided in the report.

The attached report contains a summary of the qualitative review of plant-specific AORs with respect to the post-LOCA boric acid precipitation issues identified by the NRC in Reference 1. For each participating plant, it was concluded that following a LOCA, sufficient margin exists in the methodology and assumptions to prevent the boric acid concentration in the reactor core from exceeding the solubility limit.

Please contact Paul Hijeck at (860) 731-6240 with any questions or comments regarding this information.

Very truly yours,

Paul Hijeck for

Frederick "Ted" P. Schiffley, III
Chairman, PWR Owners Group

FPS:PJH:las

Attachment

cc: G. Schukla, NRC
Analysis Subcommittee
Management Committee
D. Fink, W
R. Schomaker, AREVA
PMO

RESPONSE TO NRC REQUEST FOR JUSTIFICATION OF CURRENT OPERATION FOR POST-LOCA BORIC ACID PRECIPITATION ISSUES

A. INTRODUCTION/BACKGROUND

Three US Pressurized Water Reactor (PWR) designs (Westinghouse, CE, and B&W) use boron as a core reactivity control method and are subject to concerns regarding potential boric acid precipitation in the core for scenarios that preclude direct SI flow through the core for extended periods following a Loss-of-Coolant Accident (LOCA). All three plant designs have Emergency Core Cooling System (ECCS) features that, with or without appropriate operator action, initiate a core dilution mechanism to prevent the core region boric acid concentration from reaching the precipitation point. The common approach for demonstrating adequate boric acid dilution in a post-LOCA scenario includes simplified methods with conservative boundary conditions and assumptions. These simplified methods are used with limiting scenarios in calculations that determine the time at which appropriate operator action must be taken to initiate an active boron dilution flow path. The three US PWR designs have different ECCS designs, different procedures for preventing boric acid precipitation, and different methodologies for evaluating the potential for boric acid precipitation. Nevertheless, there are common approaches, assumptions and simplifications that have been used in virtually all PWR calculations that address the potential for boric acid precipitation. Recent Extended Power Uprate (EPU) Programs have provided the opportunity for the NRC to challenge some of these common approaches, assumptions and simplifications with regards to regulatory compliance and technical justification.

The NRC recently issued a letter (Reference 1) to the PWR Owners Group (formerly Westinghouse Owners Group) requesting that the Owners Group members confirm that they have sufficient safety margin to core cooling requirements to support continued operation. The Reference 1 letter was a follow-up to previous letter, Reference 2. A subsequent conference call with the NRC on December 7, 2005 provided some clarifications to the Reference 1 letter. The following clarifications were provided by the NRC:

- The Owners Group may provide the response to the NRC letter for the licensees. The Owners Group response letter should clearly indicate what plants are covered by the response.
- Since virtually all PWR calculations use methodology that includes some or all of the cited issues, the request is applicable to all PWRs (i.e., B&W, CE and Westinghouse NSSS designs).
- The margin of safety to core cooling is the requirement to be considered in the assessment.
- The reasonable amount of time period (i.e., 90 days) identified by the NRC is not a requirement. Flexibility in the time period is acceptable. The NRC requests the Owners Group identify when the assessment is expected to be complete.
- The NRC does not expect new quantitative analysis of boric acid precipitation as was done for Entergy Waterford 3. A qualitative assessment of margin for continued operation is acceptable. The NRC does not request a change to the plant design or licensing basis at this time.
- A high level assessment for the Reasonable Assurance of Safe Operation is acceptable provided it includes a qualitative assessment of margin in the individual plant analyses.

- The NRC clarified that issues 1-3 on the second page of Reference 1 need to be considered in the qualitative assessment of margins. Insights from the Waterford 3 analysis and compensating margins of items 1-3 on the first page of Reference 1 should be considered.
- All four issues on the second page of the Reference 1 letter need to be addressed by licensees on any future license amendments.

In response to the Reference 1 letter, the Owners Group funded a program to review the boric acid precipitation analysis-of-record (AOR) for each participating plant to support an assessment of continued safe operation in light of the post-LOCA boric acid precipitation issues identified by the NRC. It was expected that the assessment would support the stated NRC belief that plants which have relied on CENPD-254-P or similar methods have sufficient safety margin to continue operation based upon the methodology margins cited in Reference 1 and the NRC review of the Waterford 3 Long Term Cooling (LTC) analysis (Reference 3). A listing of the licensees participating in this program is provided in Table A-1.

In Section B of this report, each of the NRC stated concerns and recognized margins are discussed generically with respect to the plant design and the methodologies used to support emergency procedure action times that preclude boric acid precipitation. In Section C of this report, each of the major Nuclear Steam Supply System (NSSS) designs and supporting analyses are reviewed with respect to the NRC concerns. Margins associated with each of the major NSSS designs and supporting analyses are also discussed and conclusions are presented. Table A-3 summarizes the plant-specific AOR reviews for each plant included in this program. In order to provide additional defense-in-depth with regards to safe plant operation and regulatory compliance, Section D is a discussion of the overall safety significance of the potential for boric acid precipitation in the reactor core after a LOCA.

B. DISCUSSION OF NRC CONCERNS AND RECOGNIZED METHODOLOGY MARGINS

MIXING VOLUME AND CORE VOIDING ASSUMPTIONS

For typical plant designs (Westinghouse 2-loop upper plenum injection plants excluded) the limiting scenario for boric acid precipitation is a large cold leg (pump discharge) break where the downcomer is eventually filled and the excess SI flows out the break. The SI flow into the core region is limited to that quantity boiled off in the core to remove the decay heat. The steam generated in the core travels around the intact hot leg (s) (or through the internals reactor vessel vent valves (RVVVs) in the B&W-designed plants) to exit the break. Boron left behind accumulates in the core region and the boron concentration in the core region increases. The calculated rate of increase in boron concentration in the core region after a LOCA is directly affected by the assumed liquid volume. During this time, the core and upper plenum are filled with a two-phase mixture whose liquid content is dependent on the degree of voiding in the core and upper plenum region. The degree of voiding is controlled by the core decay heat, and Reactor Coolant System (RCS) pressure, and the pressure drop around the loop or through the RVVVs as it affects the hydrostatic balance between the downcomer head and the collapsed liquid level in the core. At low RCS pressures and high decay heat levels, the boiling in the core is vigorous, and the volume of actual liquid in the core region is smaller. As the decay heat drops off, the boiling becomes less vigorous and more liquid is retained in the core region.

Westinghouse US 2-loop plants differ from typical PWR designs in that they utilize a low pressure upper plenum safety injection (or UPI). For these plants, the limiting large break LOCA boric acid precipitation scenario is a hot leg break where the cold leg high pressure SI may be

terminated at or prior to sump recirculation. This scenario is relevant only with the very conservative assumption that all UPI flow in excess of core boiloff bypasses the core region and flows directly out the break (i.e. no mixing in the core and upper plenum).

Throughout the US PWR fleet, it is common to find analyses that use simplifying assumptions in determining a liquid mixing volume used in boric acid precipitation calculations. Such assumptions typically ignore the effect of voiding in the core while conservatively selecting the mixing volume boundary. A conservative mixing volume boundary might ignore regions that would be expected to participate, at least partially in the mixing process.

As discussed later in this report, concerns over methodologies that do not explicitly consider voiding can be addressed by recognizing conservative liquid mixing volume assumptions and other conservatisms in the calculations.

SYSTEM EFFECTS (MIXING VOLUME VERSUS TIME)

This NRC concern addresses the core/upper plenum liquid mixing volume early in the post-LOCA transient where decay heat and the associated steaming rates are high. During this time, the loop or RVVV pressure drop is relatively high and core collapsed liquid level, which is in hydrostatic balance with a filled downcomer, is low. If the liquid volume in the core and upper plenum is sufficiently small there is a greater potential for boric acid precipitation in the core. As the decay heat decreases during the transient, the mixing volume increases due to the mixture level swell into the upper plenum and a decrease in the average void fraction in the mixture region.

Throughout the US PWR fleet, it is common to find analyses that ignore the loop pressure drop effects in determining a liquid mixing volume used in boric acid precipitation calculations. As with core voiding, this simplification is justified by conservatively selecting the mixing volume boundary.

As discussed later in this report, concerns over methodologies that do not model loop pressure drop can be addressed by conservative liquid mixing volume assumptions and other conservatisms in the calculations.

BORIC ACID SOLUBILITY LIMIT

The assumed boric acid solubility limit directly affects the acceptability of passive or active post-LOCA core dilution mechanisms. Throughout the US PWR fleet, it is common for analyses to use boric acid solubility limit values consistent with data reported in Reference 4 or Reference 5. Boric acid solubility data from Reference 4 is shown in Table A-2, which is very similar to the Reference 5 values. For large break LOCA scenarios, it is most common for analyses to use the atmospheric pressure boric acid solubility limit value. Analyses may include some margin for uncertainties in the solubility limit and other uncertainties to account for factors such as temperature variations within the core region.

Some analyses credit the higher boric acid solubility limit associated with higher core region mixture temperatures based on higher RCS pressures for SBLOCAs or elevated containment pressure for any LOCA. It is also possible that some analyses credit the higher boric acid solubility limit that would result from containment sump additives used to control sump pH. The effects of containment sump pH control additives are discussed in more detail later in this section.

DECAY HEAT

A variety of decay heat bases have been used in the post-LOCA boric acid precipitation analyses of the US PWR fleet. For some plants, the analyses were treated as technical support for EOP

guidance and consequently were considered to be outside of 10CFR50.46 Appendix K requirements. In other cases, the use of non-Appendix K decay heat was justified in light of other conservatisms in the methodology. Still in other cases, a modified Appendix K decay heat was used on the basis of the presumed intent of the 10CFR50.46 Appendix K decay heat standard as it would be applied to transients greater than 1000 seconds.

INCREASE IN BORIC ACID SOLUBILITY LIMIT DUE TO CONTAINMENT SUMP PH ADDITIVES

All plants in the US PWR fleet utilize alkaline additives (i.e. TSP, NaOH, or sodium tetraborate) to control the pH of the water in the containment sump after a LOCA. A list of buffering agents is provided in Appendix A. Containment sump pH control additives increase the boric acid solubility limit by increasing the ionized fraction of boric acid solution.

Few US PWR fleet post-LOCA boric acid precipitation analyses credit the increased solubility limit that may results from containment sump pH additives. One analysis that has credited the effects of trisodium phosphate (TSP) is the Waterford 3 Extended Power Uprate Long Term Cooling analysis, discussed in Reference 6. Reference 6 cites tests indicating a boric acid solubility limit of >36% for the expected ternary solution of water, boric acid, and TSP undergoing boiling at saturation under atmospheric pressure conditions. Recently, RAI (Request for Additional Information) responses for a long term cooling analysis under NRC review (Reference 7), cites testing that shows a boric acid solubility limit of >48% for ternary solution of water, boric acid, and NaOH undergoing boiling at saturation under atmospheric pressure conditions. No specific boric acid solubility limit for a ternary solution of water, boric acid, and sodium tetraborate can be cited. However, since all pH control additives will increase the ionized fraction of boric acid, all of the pH control additives would be expected to increase the boric acid solubility of the solution. This expectation is supported by the preliminary chemistry testing of boric acid, water, and sump pH buffering agents being performed for the PWR Owners Group for the GSI-191.

UN-CREDITED MIXING VOLUME

Virtually all of the post-LOCA boric acid precipitation calculations that support the US PWR fleet use simplifying conservative assumptions with regards to mixing volume. Since the assumed mixing volume directly affects the calculated rate of increase of boric acid in the core and upper plenum region simplified conservative assumptions represent a source of margin in the predicted time to reach the solubility limit. Some of the common reactor vessel volumes that may have been excluded from the mixing volume include;

- Lower plenum volume
- Volume above the core region that is above the bottom of the hot leg elevation
- Volume of the liquid in the hot leg
- Core bypass volume (thimble tubes/barrel baffle region, etc.)
- Downcomer volume

The lower plenum volume in particular represents a significant source of margin since the lower plenum volume is relatively large and would be essentially all liquid. Mixing in the lower plenum is driven by liquid density differences due to boric acid concentration gradients. Participation of the lower plenum in the post-LOCA vessel mixing process was evidenced in MHI

BACCHUS tests (Reference 8). For the Waterford 3 Expended Power Uprate as well as pending Beaver Valley and Ginna EPU Programs, the NRC has accepted the assumption of mixing in 50% of the lower plenum.

INCREASE IN BORIC ACID SOLUBILITY AT PRESSURES ABOVE ATMOSPHERIC PRESSURE

As mentioned earlier, throughout the US PWR fleet for large break LOCA scenarios, it is most common for analyses to use the atmospheric pressure boric acid solubility limit value. At containment pressures above atmospheric, the boiling temperature of the boric acid and water solution increases and the solubility limit increases correspondingly. For an assumed containment pressure of 20 psia, the boiling point of pure water is 228°F and the boiling point of a boric acid and water solution is even higher. As indicated in Reference 4, the solubility limit of a 228°F boric acid and water solution is >32 wt%. Furthermore, any pressurization of the reactor vessel above the containment pressure (due to the pressure drop around the loop(s)) would increase the expected temperature of the core region mixture and would add even more margin to the boric acid precipitation point.

C. AOR REVIEWS AND CONCLUSIONS

Table A-3 summarizes the results of the AOR reviews for Westinghouse, CE, and B&W NSSS plant designs, respectively. A discussion of each of the evaluations for each of the designs follows.

Westinghouse NSSS Design Plants

The Westinghouse methodology for demonstrating the preclusion of boric acid precipitation was first documented in Reference 9. The methodology description in Reference 9 is non-specific in many respects and a number of generic analyses results are presented. Generic calculation input assumptions were used and alternate calculations performed (for example, calculations both with and without lower plenum volume). Reference 9 also cited pending calculations to demonstrate the existence of circulation flow patterns in the core after the hot leg safety injection flow path was established. These results of these calculations were later reported in Reference 10.

Most of the boric acid precipitation boric analyses performed by Westinghouse for Westinghouse design plants follow the methodology in Reference 9. Westinghouse has also used the Reference 9 methodology for one non-Westinghouse plant, Fort Calhoun. Therefore, all subsequent discussion of Westinghouse plant AORs also applies to this plant as well.

For each Westinghouse NSSS design plants (and Fort Calhoun) the HLSO AOR was reviewed and the boric acid concentration at HLSO time was established. The boric acid concentration at HLSO time was then adjusted to account for the NRC Issues #1 through #4 from Reference 1. It should be noted that Appendix K decay heat (NRC Issue #4) was considered even though the most plants were licensed with calculations that used realistic decay heat. The boric acid concentration at HLSO time was then adjusted further by crediting margins cited by the NRC in Reference 1 (effect of sump additive, additional mixing volume, containment overpressure) along with previously unclaimed solubility limit margins.

Although nearly all AORs used a non-Appendix K decay heat, nearly all AORs used a conservative boric acid solubility limit of 23.53 wt%. These 2 effects approximately offset each other. Similarly, nearly all AORs did not explicitly include core voiding and nearly all AORs used a conservative mixing volume that did not include the lower plenum, barrel/baffle region,

thimble tubes, nor volume above the bottom of the hot leg elevation. Taken all together, voiding effects and un-credited volumes approximately offset each other. The nature of the compensating effects of the assumptions on decay heat, solubility limit, voiding and mixing volume was demonstrated in re-analyses performed for RAIs for the Beaver EPU program (Reference 7). For nearly all plants included in this evaluation, neither containment overpressure nor the benefit of sump additives was needed to show that the boric acid concentration at HLSO time was below the solubility limit.

Westinghouse 2-loop plants with UPI injection are non-standard with respect to boric acid precipitation since large cold leg breaks do not present a boric acid precipitation concern since the UPI flow provides immediate core dilution flow. For hot leg breaks, the methodology described in Reference 9 has been applied to determine the time to restore simultaneous injection to the cold leg injection should it be terminated after aligning to sump recirculation Alternately, credit for UPI/core region mixing (discussed in Reference 10) has be used to address the potential for boric acid buildup for this condition. The latter position was accepted recently by the NRC (Reference 11) and was recognized as reasonable at recent ARCS meetings (References 12 and 13). Since UPI flow provides flushing flow for large cold leg breaks, and since UPI/core region mixing prevents boric acid precipitation for large hot leg breaks, the AOR assumptions listed in Table A-3 have been identified as "Not Applicable" for Westinghouse 2-loop UPI plants.

As described earlier, the Westinghouse NSSS plants reviewed in evaluation were licensed on the methodology described in Reference 9. This methodology treated SBLOCAs as non-limiting due to the higher boric acid solubility limits for saturation temperatures at pressures significantly above atmospheric (refer to Table A-2). Since this approach was part of the licensing basis, and since SBLOCA scenarios were not specifically mentioned by the NRC in Reference 1, SBLOCAs and associated cooldown scenarios were not part of the evaluations performed for this program.

In summary, for all Westinghouse NSSS plants (and Fort Calhoun), evaluations were performed to adjust the boric acid concentration at HLSO time to address the NRC concerns in Reference 1. For all plants, the adjusted boric acid concentration at HLSO time was below the expected boric acid solubility limit.

CE NSSS Design Plants

The methodology for calculating the prevention of boric acid precipitation for CE PWRs is described in topical report CENPD-254-P-A (Reference 14). In particular, the analysis procedure is described in Section 3.2 of CENPD-254-P-A and the results of a sample analysis are presented in Section 4.1. The analysis uses the BORON computer code to calculate the boric acid concentration in the core region as a function of time following a LBLOCA. BORON is described in Appendix C of CENPD-254-P-A. The analysis is performed for a LBLOCA located in a RCP discharge leg, the limiting break location for the prevention of boric acid precipitation in the core region. In addition to describing the boric acid precipitation analysis methodology for LBLOCAs, CENPD-254-P-A also describes the analysis methodology for demonstrating long-term decay heat removal for SBLOCAs.

The current post-LOCA boric acid precipitation analyses for the CE PWRs (excluding Fort Calhoun) were evaluated with respect to the impact of the issues identified in Reference 1 on the boric acid concentration in the core region at the time of switchover to simultaneous hot and cold leg injection. The evaluations also considered the impact of relaxing conservative features in the current analyses, similar to those credited in the Waterford 3 Extended Power Uprate boric acid precipitation analysis and identified in Reference 1. The evaluations for some of the CE PWRs included running the BORON computer code to assess the net impact of the Reference 1 issues

and the compensating conservatisms on the results of the current analyses. The evaluations demonstrated that the maximum boric acid concentration in the core region remained below the solubility limit for each plant after consideration of the issues identified in Reference 1 and the relaxation of compensating conservatisms in the current analyses.

B&W NSSS Design Plants

The AREVA methodology for demonstrating that boric acid precipitation does not occur is described in References 15 and 16. Some of the B&W-designed plants continue to use the generic analyses described in these references, while others have used the methods developed for these generic analyses to address plant specific boron dilution methods or limitations on the dilution methods (e.g. plant specific auxiliary pressurizer spray flows, protection of the sump screens, prevention of potential water-hammer scenarios in the decay heat piping, challenges to NPSH limits for LPI pumps, hot and cold fluid mixing limits, prevention of boric acid precipitation inside the decay heat cooler, etc.).

A review of the current post-LOCA boric acid precipitation methods and calculations for the B&W-designed plants (excluding Oconee) with respect to the NRC concerns identified in Reference 1 was performed. Each plant was found to have an overall conservative method to assure that the boric acid concentration in the core would not exceed the solubility limit following a LOCA. The degree of conservatism varied from plant to plant although most plants included the specific conservative methods suggested by the NRC. In cases where all the specific conservatisms were not included there may have been different credits or conservatisms, that taken as a whole, support ample conservatism to assure that the requirements of 10 CFR 50.46 are met.

D. ASSESSMENT OF THE SAFETY SIGNIFICANCE OF THE POTENTIAL FOR BORIC ACID PRECIPITATION IN THE REACTOR VESSEL AFTER A LOCA

The purpose of this section is to discuss the overall safety significance of the potential for boric acid precipitation in the reactor vessel after a LOCA. This discussion supplements the evaluations presented earlier and provides defense-in-depth for support of current plant safe operation. The potential risk of boric acid precipitation is the effect on core cooling and the increase in CDF that may result from an incidence of post-LOCA boric acid precipitation. With this perspective in mind, the boric acid precipitation concerns cited in References 1 and 2 do not represent a significant safety for the following reasons.

- There is low probability of a LBLOCA where conditions leading to significant boric acid accumulation may be encountered.
- Some of the transient behavior concerns discussed in Reference 2, if incorporated into boric acid precipitation analyses, would have a beneficial effect on the results. These include; liquid entrainment around the loop, boron carryover in the steam, containment overpressure, and mixing in regions outside the core, upper plenum and portions of the lower plenum. Modeling realistic subcooling for lower plenum fluid would provide significant benefits in terms of reduced boiloff rates. Loop seal replugging, were it to occur, could cause temporary core level depression which would force additional fluid mixing in the lower plenum thus reducing boron buildup rates. Given the size of the steam generators and the reduced steam flow rates through the RCS post-LOCA, it would take substantial depositions of boron to have any significant effect on pressure in the upper

plenum, and boron deposited in the steam generator would not be available to precipitate in the core.

- If best estimate or realistic assumptions are used, the predicted boric acid precipitation time would be significantly longer than designated EOP action times that address boric acid precipitation concerns. The most significant realistic assumptions are best estimate decay heat, nominal boron concentrations and water source volume assumptions, and some level of liquid entrainment out of the core early in the transient.
- Observations from a test facility representative of a typical Westinghouse 3-loop PWR indicate that boric acid precipitation would not occur for at least 24 hours after a large break LOCA (Reference 8). This indicates a boric acid buildup well below that calculated with methodologies in question.
- In the event that boric acid precipitation should occur, it is unlikely that core cooling would be compromised.
    - It is expected that the boric acid precipitate would tend to plate out on the colder structures, accumulate in the lower plenum, or would collect at the top of liquid mixture level. This tendency was noted in tests documented in Reference 17.
    - Any boric acid precipitate would go back into solution once the core dilution flow is initiated.
    - The ultimate ability of boric acid to insulate the fuel rods is limited since orthoboric acid precipitate has a melting point of 340°F.
    - It would take significant boric acid precipitate to totally block water from getting to the core. For example, for a typical Westinghouse 3-loop PWR, if <u>all</u> of the boric acid in the expected decay heat core boiloff were to precipitate after 4 hours after the LOCA the total precipitate for the next 4 hours would be displace less than 58 cubic feet, or less that 5% of the free volume of the total core and lower plenum.

E. REFERENCES

1. NRC letter dated November 23, 2005, D.S. Collins to G. C. Bischoff, "Suspension of NRC Approval for use of Westinghouse Topical Report CENPD-254-P, Post LOCA Long term Cooling Model, Due to Discovery of Non-Conservative modeling Assumptions During Calculation Audit".
2. NRC letter dated August 1, 2005, R. A. Gramm to J. A. Gresham, "Suspension of NRC Approval for Use of Westinghouse Topical Report CENPD-254-P, 'Post LOCA Long-term Cooling Model' Due to Discovery of Non-conservative Modeling Assumptions During Calculations Audit"
3. Letter, Kalyanam to (NRC) to J. E. Venable (Entergy), WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT RE: EXTENDED POWER UPRATE (TAC NO. MC1355), April 15, 2005.
4. Cohen, P., 1969, Water Coolant Technology of Power Reactors, Chapter 6, "Chemical Shim Control and pH Effect," ANS-USEC Monograph.
5. Kracek, Morey, and Mervin, "The System, Water – Boron Oxide," American Chemical Society Paper, April 19, 1938
6. Entergy Letter to NRC, W3F1-2005-0012, "Supplement to Amendment Request NPF 38 249, Extended Power Uprate, Waterford Steam Electric Station, Unit 3," February 16, 2005.

7. L-05-112, Letter from First Energy Nuclear Operating Company to USNRC, "Beaver Valley Power Station Unit Nos. 1 and 2, BV-1 Docket No. 50-334, License NO. DPR-66, BV-2 Docket No. 50-412, License No. NPF-73, Responses to a Request for Additional Information in Support of License Amendment Request Nos. 302 and 173," July 8, 2005.
8. WCAP-16317-P, "Review and Evaluation of MHI BACCHUS PWR Vessel Mixing Tests," November 2004.
9. CLC-NS-309, Letter from C. L. Caso (Westinghouse) to T. N. Novak (NRC), 04-01-75.
10. JOC-NS-354, Letter from J. O. Cermak (Westinghouse) to T. N. Novak (NRC), 07-23-75.
11. NRC Letter, John G. Lamb (NRC) to Mr. Thomas Coutu (Kewaunee), KEWAUNEE NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT REGARDING STRETCH POWER UPRATE (TAC NO. MB9031), Enclosure 2, Kewaunee Nuclear Power Plant, Safety Evaluation for Amendment No. 172 Stretch Power Uprate, February 27. 2004.
12. Nuclear Regulatory Commission, Advisory Committee on Nuclear Safeguards Subcommittee on Power Uprates Meeting, April 27, 2006.
13. Nuclear Regulatory Commission, Advisory Committee on Nuclear Safeguards 532nd Meeting, May 04, 2006.
14. CENPD-254-P-A, "Post-LOCA Long-Term Cooling Model.
15. AREVA NP Document 51-1266113-00, "Post-LOCA Boron Concentration Management," April 1997.
16. AREVA NP Document 51-1206351-00, "Long Term Boron Dilution", January 1992.
17 CE Report LOCA-75-127, "Post LOCA Boric Acid Mixing Experiment," 10-06-75.

Table A-1 Participating Plants	
D. C. Cook 1	VC Summer
D. C. Cook 2	Wolf Creek
Braidwood 1	Diablo Canyon 1
Braidwood 2	Diablo Canyon 2
Byron 1	South Texas 1
Byron 2	South Texas 2
Callaway 1	Crystal River 3
Palo Verde 1	Sequoyah 1
Palo Verde 2	Sequoyah 2
Palo Verde 3	Watts Bar 1
Turkey Point 3	Fort Calhoun
Turkey Point 4	Farley 1
St. Lucie 1	Farley 2
St. Lucie 2	Vogtle 1
Kewaunee 1	Vogtle 2
Millstone 2	Calvert Cliffs 1
Millstone 3	Calvert Cliffs 2
North Anna 1	Davis-Besse 1
North Anna 2	Ginna 1
Surry 1	Beaver Valley 1
Surry 2	Beaver Valley 2
HB Robinson 2	Waterford 3
Shearon Harris 1	Point Beach 1
Palisades 1	Point Beach 2
Prairie Island 1	Indian Point 2
Prairie Island 2	Indian Point 3
Seabrook 1	ANO 1
San Onofre 2	ANO 2
San Onofre 3	Salem 1
TMI 1	Salem 2
Comanche. Peak 1	
Comanche. Peak 2	

Table A-2 Solubility Limits of H_3BO_3 in H_2O[Cohen (1969)]		
	Boric Acid Solubility	
Temperature, °C (°F)	Weight % Boric Acid	PPM Boron [1]
P = 1 Atmosphere		
0 (32)	2.70	3,497
5 (41)	3.14	5,490
10 (50)	3.51	6,137
15 (59)	4.17	7,291
20 (68)	4.65	8,130
25 (77)	5.43	9,494
30 (86)	6.34	11,085
35 (95)	7.19	12,571
40 (104)	8.17	14,284
45 (113)	9.32	16,295
50 (122)	10.32	18,043
55 (131)	11.54	20,177
60 (140)	12.97	22,677
65 (149)	14.42	25,212
70 (158)	15.75	27,537
75 (167)	17.41	30,440
80 (176)	19.06	33,325
85 (185)	21.01	36,734
90 (194)	23.27	40,685
95 (203)	25.22	44,095
100 (212)	27.53	48,133
103.3 (217.9)	29.27 [2]	51,176
$P = P_{SAT}$		
107.8 (226.0)	31.47	55,022
117.1 (242.8)	36.69	64,149
126.7 (260.1)	42.34	74,027
136.3 (277.3)	48.80	85,322
143.3 (289.9)	54.80	95,812
151.5 (304.7)	62.22	108,785
159.4 (318.9)	70.70	123,612
171 (339.8) = Congruent Melting of H_3BO_3		

Table A-3 Plant AOR Review Summary

Group	# of Plants	Plant Design	Action Time (hr) [a]	Core Voiding Included?	System Effects Included?	LBLOCA AOR BA Conc At EOP Action Time [b] (BA wt%)	Decay Heat [c,d]	Lower Plenum Volume Credited (%)	Volume Above Bottom of HL Credited?	Effect of Sump Additives Credited?	Containment Overpressure Credited?
					----- NRC Issues from Nov. 23, 2005 Letter Items (1) thru (4) -----|				------ Margins from Nov. 23, 2005 Letter Items 1) thru 4) ------|		
B-1	3	B&W	4.94 - 7.56	YES	YES	<24.20	Appendix K	0	In UP/not in HL	NO	NO
B-2	1	B&W	24	No	No	<28.7	71 ANS	100	NO	NO	NO
C-1	4	CE	3 - 10	NO	NO	<27.6	71 ANS x 1.2/1.1	100	NO	NO	NO
C-2	4	CE	3 - 6.5	NO	NO	<32	71 ANS x 1.2/1.1	100	NO	NO	YES
C-3	2	CE	20	NO	NO	<32	79 ANS + 2 Sigma	100	NO	NO	YES
C-4	1	CE	13	NO	NO	<27.6	71 ANS x 1.2/1.1	100	NO	NO	NO
C-5	1	CE	6	NO	NO	<27.6	71 ANS x 1.2/1.1	0	NO	NO	NO
C-6	1	CE	3	YES	NO	<36	71 ANS x 1.2/1.1	50	UP to top of HL	YES	NO
C-7	1	CE	8.5	NO	NO	< 23.53	Appendix K	0	NO	NO	NO
W-1	24	W	5.5 - 14	NO	NO	< 23.53	71 ANS Finite	0	NO	NO	NO
W-2	8	W	3 - 6.5	NO	NO	< 23.53	Appendix K	0	NO	NO	NO
W-3	2	W	3	NO	NO	<< 23.53	71 ANS Finite	0	NO	NO	NO
W-4	2	W	5	NO	NO	< 23.53	79 ANS x 1.2	0	NO	NO	YES
W-5	2	W	8	NO	NO	< 23.53	79 ANS	0	NO	NO	YES
W-6 [e]	1	W-UPI	20	NA	NA	NA	NA	NA	NA	NA	NA
W-7 [e]	3	W-UPI	NA	NA	NA	NA	NA	NA	NA	NA	NA
W-8 [e]	2	W-UPI	14	NA	NA	NA	NA	NA	NA	NA	NA

NA = Not Applicable

[a] EOP Action Time is latest hot leg switchover time, time to switch to simultaneous injection, or other actions to initiation core dilution.

[b] Large Break LOCA scenario unless noted otherwise.

[c] Decay heat is for infinite irradiation of U235 unless otherwise noted.

[d] 1.2/1.1 indicates 20% uncertainty out to 1000 sec., 10% beyond 1000 sec.

[e] For W 2-Loop UPI plants, UPI flow provides flushing flow for cold leg breaks and UPI/core region mixing prevents boric acid precipitation for hot leg breaks.