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

April 27, 2011

Mr. Ashok S. Bhatnagar
Senior Vice President
Nuclear Generation Development
and Construction
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 – AUDIT REPORT OF WESTINGHOUSE DOCUMENTS RELATING TO FINAL SAFETY ANALYSIS REPORT ACCIDENT ANALYSES (TAC NO. ME4620)

Dear Mr. Bhatnagar:

The U.S. Nuclear Regulatory Commission (NRC) staff performed an audit of Westinghouse documentation that supports Tennessee Valley Authority's license application for Watts Bar Nuclear Plant, Unit 2. The audit took place on March 15, 2011, at the Westinghouse office located in Rockville, Maryland. Enclosed is the audit summary report prepared by the NRC staff.

If you should have any questions, please contact me at 301-415-2048.

Sincerely,

Justin C. Poole, Project Manager
Watts Bar Special Projects Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-391

Enclosure:
Audit Summary

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REGULATORY AUDIT SUMMARY OF THE

FINAL SAFETY ANALYSIS REPORT SECTION 15 ACCIDENT ANALYSES

WATTS BAR NUCLEAR PLANT, UNIT 2

By letter to the Nuclear Regulatory Commission (NRC) dated January 29, 2008, the Tennessee Valley Authority (TVA), the applicant for an operating license for the Watts Bar Nuclear Plant (WBN) Unit 2, submitted a description of the regulatory framework for the completion of licensing activities for WBN Unit 2. Consequently, the NRC staff resumed its review of TVA's license application.

The audit was conducted on March 15, 2011, at the Westinghouse office in Rockville, Maryland.

The audit team consisted of:

Samuel Miranda, Division of Safety Systems, Reactor Systems Branch
Benjamin Parks, Division of Safety Systems, Reactor Systems Branch
Leonard Ward, Division of Safety Systems, Nuclear Performance Code Review Branch
Paul Clifford, Division of Safety Systems
Pat Milano, Division of Operating Reactor Licensing, Watts Bar Special Projects Branch
Justin Poole, Division of Operating Reactor Licensing, Watts Bar Special Projects Branch

TVA participants:

Frank Koontz
Gordon Arent

Westinghouse participants:

Chris Morgan
Alan Macdonald
Ryan Rossman
Chris McHugh
Dave Chapman
Dave Fink
Dan Utley

The purpose of this audit was to allow the NRC staff (1) to read additional material that might be proprietary in nature, (2) to discuss the content of said material with cognizant technical experts, and (3) to minimize the need to write further requests for additional information (RAIs).

The audit focused upon the following topics:

A. Steamline Break Analyses and Methods
B. Mass Addition Events
C. Reactor Coolant System (RCS) and main steam system overpressure analysis
D. Loss-of-Coolant Accident (LOCA)

Basically, the audit produced more questions than answers. This audit report will summarize the staff's questions and their bases in order to provide information that may be useful in developing relevant, definitive responses for the completion of staff's review. For this purpose, reference is made to prior RAIs, the licensee's responses, and the staff's understanding of said responses. In some cases, additional, follow-up questions are posed, which are intended to address the staff's underlying concerns that have not yet been settled.

A. Steamline Break Analyses and Methods

A.1 What is the limiting steamline break case?

Two steamline break analyses are described in Final Safety Analysis Report (FSAR) Section15.4.2.1:

(a) Complete severance of a pipe, with the plant initially at no load conditions, full reactor coolant flow with offsite power available, and

(b) Case (a) above with loss of offsite power. Loss of offsite power results in coolant pump coastdown.

There is some confusion as to which of these cases is considered to be the limiting case. In the FSAR, it is stated that, "The limiting statepoints for the two cases are presented in Table 15.4-7." However, Table 15.4-7, "Limiting Core Parameters Used in Steam Break DNB [Departure from Nucleate Boiling] Analysis," lists just one statepoint, for case (a). Yet the FSAR states, "Case b results in a more limiting return to power than Case a."

Hence the staff's RAI and applicant's response:

Q. Identify the case (a or b) from which the statepoint listed in Table 15.4-7, "Limiting Core Parameters Used in Steam Break DNB Analysis," is taken.

A. The limiting statepoint is from Case a (with offsite power available). This case resulted in the limiting combination of thermal-hydraulic conditions (heat flux, inlet temperature, pressure, flow, reactivity, etc). The low flow case is non-limiting as was discussed in the licensing of the SLB [steamline break] topical report; see the USNRC Question 222.11 and Westinghouse response (WCAP-9226-P-A, Rev. 1, Section D). The response identified that using an open channel coupled-code was necessary for evaluation of the low flow case and that the low flow case DNBR [DNB ratio] is bounded by the full flow case. Thus, although the plant response to a steam line break event with a loss of offsite power is presented in the Unit 2 FSAR, the case itself has been shown to be non-limiting with respect to DNBR.

Here is the question and answer set that is cited in the applicant's response (above):

Q) 222.11

A recent CE [Combustion Engineering] paper entitled "Design Analysis Using Coupled Neutronic and Thermal Hydraulic Models" by S. G. Wagner et al. was presented at the

Topical Meeting on Advances in Reactor Physics in Gatlenburg, Tennessee (April 1978). This paper presented an evaluation of steam line break analyses using a 3-dimensional coupled thermal-hydraulic neutronic code. Compare the results of this paper with the results of WCAP-9226. For example, the paper shows the cross flow in fuel bundles (open channels) inserts additional reactivity. Since WCAP-9226 assumed closed fuel channels instead of open fuel channels, discuss how this effect was considered in WCAP-9226.

A) 222.11

Westinghouse recognizes the existence of cross flow phenomena during steamline break conditions and has considered their effect on the results of the analysis. We have concluded that:

1. For steamline break with offsite power available (full reactor coolant flow), the closed channel model is very accurate.

2. Using DNBR as a basis (rather than reactivity, as the CE paper does), the conclusion stated in WCAP-9226 Section 3.1.1.14 concerning the steamline break without offsite power available, i.e., low RCS flow, can be substantiated with an open channel model, where the effects of cross flow is considered.

The use of closed channel calculations for the more limiting full flow cases will continue to be the basis for Westinghouse licensing calculations.

During the audit, the NRC staff indicated that the answer is not understood. Westinghouse agreed to develop another, simpler response. Here are some of the staff's concerns regarding this question:

a. The term "limiting" is not well-defined. Is a case more "limiting" than another due to higher reactivity, or higher post-trip power level, lower boron concentration, lower RCS flow, or lower DNBR, or lower pressure, or a combination of these parameters? If it's a combination, then describe how the combination is determined, justified, and applied.

b. Westinghouse provided a computer printout of statepoints. A statepoint for every time step of the steamline break analysis is transmitted to the core designers for a reactivity check, before the minimum DNBR is determined. These statepoints were taken from the results of the Case (a) steamline break analysis (with offsite power). There were no statepoints from the results of the Case (b) steamline break analysis (without offsite power). Westinghouse stated, in an RAI response, that the DNBR was not evaluated for the steamline break without offsite power case. This seems to be consistent with Westinghouse's conclusion that steamline breaks without offsite power are always less limiting than steamline break cases with offsite power (i.e., with respect to minimum DNBR).

This practice may be the implementation of a conclusion reached in WCAP-9226 § 3.1.1.14, that is based upon the results of steamline break sensitivity studies. WCAP-9226 states, "A comparison of Figure 3.1-39 with Figure 3.1-6 shows that both the core power and core flow are lower for the case without offsite power." Here are Figures 3.1-39 and 3.1-6 from WCAP-9226:

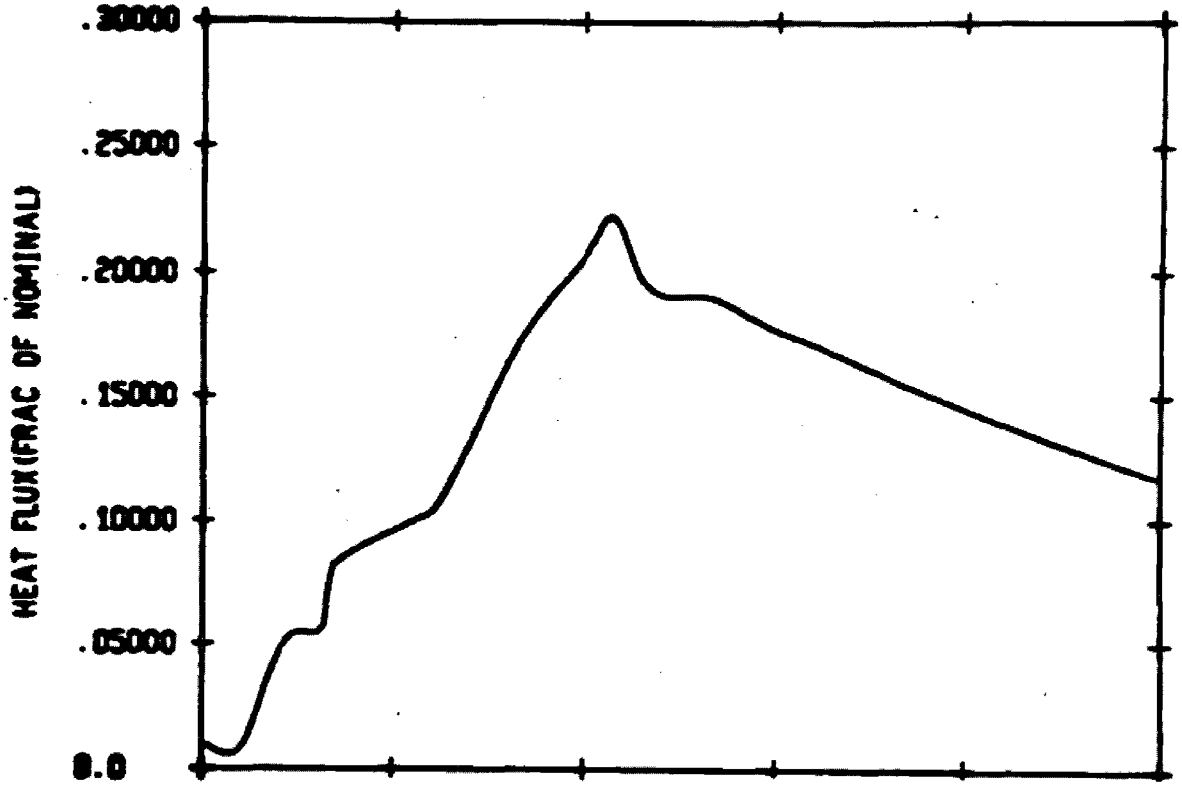


Figure 3.1-6: 4.6 ft^2 break with offsite power (Reference Case, 0 – 100 s)

Notes: (1) Figure 3.1-6 does not include a plot of core flow vs. time, since full flow is maintained, and (2) in 1978, the 4.6 ft^2 break was the largest possible steamline break (today, it is 1.4 ft^2).

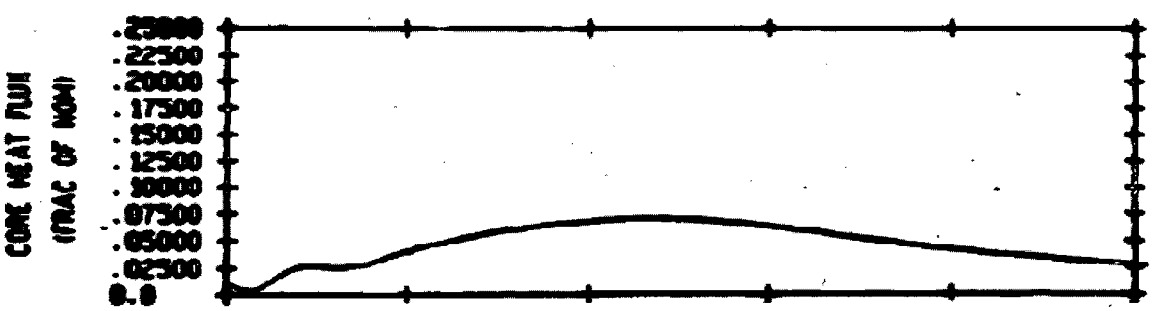


Figure 3.1-39: 4.6 ft^2 break without offsite power (0 – 100 s)

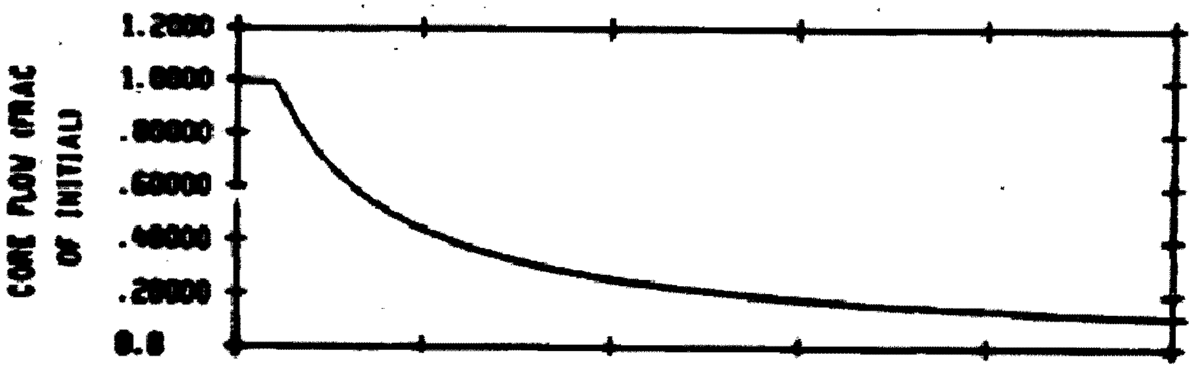


Figure 3.1-39 (cont'd) 4.6 ft^2 break without offsite power (0 – 100 s)

For WBN Unit 2, the peak post-trip power level for the steamline break analysis with offsite power occurs 1 minute after the break and reaches 1.6 percent of nominal. For the steamline break analysis without offsite power, the peak post-trip power level occurs 2 to 3 minutes after the break and reaches a higher level, as high as 7 percent, and remains at that level for a much longer period. This peak power level is based upon a very poor quality plot of nuclear power. (A plot of core power was not submitted.) The core power is scaled from the nuclear/core power ratio of the 2011 WBN Unit 2 steamline break with offsite power case. The RCS flow rate, at the time of peak power (2 to 3 minutes), would be lower, since it would have more time to decay.

The staff questions whether the WCAP-9226 conclusion, that steamline breaks without offsite power will always be less limiting than steamline breaks with offsite power, would apply to WBN Unit 2, since it is based upon the statement that the peak post-trip power levels for the steamline break analyses without offsite power will always be lower than the peak post-trip power levels for the steamline break analyses with offsite power. The reverse is seen in the WBN Unit 2 steamline break analysis results. This suggests the possibility that the WBN Unit 2 steamline break without offsite power could be the limiting case. Westinghouse has performed the steamline break transient analysis for this case; but hasn't determined its resultant minimum DNBR.

Furthermore, Westinghouse's claim that analyses of steamline breaks without offsite power, using an open channel model with cross flow, "substantiate" the conclusion in WCAP-9226 is not substantiated.

Followup questions:

1. Provide a legible plot of the core heat flux transient predicted by the 1.4 ft^2 steamline break without offsite power analysis, extending to at least 10 minutes.

2. Provide the limiting statepoint and DNBR evaluation for the 1.4 ft^2 steamline break without offsite power analysis. Compare the minimum DNBR for this case to the minimum DNBR from the case that assumes offsite power is available, and explain the difference.

3. Explain the conclusion in WCAP-9226, that the full flow cases are more limiting, and show that this conclusion applies to the results of the WBN Unit 2 steamline break analyses.

4. Address the staff's question, raised during the audit, regarding fuel rod mechanical design rod internal pressure (i.e., clad liftoff) concerns during the steamline break event down to the accumulator injection setpressure.

5. What, if any, actions would plant operators initiate if the moderator temperature coefficient (MTC) surveillance measurement was within the most-negative allowable MTC, but beyond the cycle-specific moderator density coefficients used in the steamline break analysis?

A.2 WBN Unit 2 steamline break analysis reactivity coefficients

The applicant's analyses of the major steamline break, reported in Chapter 15 of the FSAR, indicate that the 1.4 ft^2 break, with offsite power available, produces a post-trip peak power level of 1.6 percent of nominal. In other plants of WBN Unit 2's size and design, the same steamline break size, with offsite power available, produces post-trip peak power levels that are two to six times greater. The applicant's fuel vendor, Westinghouse, attributed this to the effect of relatively favorable moderator reactivity coefficients. Since these moderator reactivity coefficients seem to be having a profound effect upon the WBN Unit 2 analyses results, the staff seeks further information regarding how they're determined, verified, and applied.

At the audit, Westinghouse began a historical review of the development of reactivity coefficients, to evaluate how and why the reactivity coefficients have changed to their current values. In addition, the staff questioned the use of predicted cycle-specific moderator density coefficients and their relationship to technical specifications/core operating limits report (TS/COLR) most-negative MTC limits. In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36(c), the most-negative MTC limits (and surveillance requirements) in the TS/COLR should be linked directly with plant safety analyses, specifically main steamline break. The staff asked what, if any, actions would plant operators initiate if the MTC surveillance measurement was within the most-negative allowable MTC, but beyond the cycle-specific moderator density coefficients used in the steamline break analysis.

The staff expects to receive the results of this review, when it's completed, and an explanation detailing how the reactivity coefficients have changed, why they changed, and how the changes have been verified (e.g., by plant measurements). This information should also describe the derivation and development of stuck rod reactivity coefficients, used in steamline break analyses, from the design reactivity coefficients.

Basically, the staff wants to understand why a given cooldown rate (caused by a 1.4 ft^2 steamline break) has less of an effect upon WBN Unit 2 than it does on other plants of similar design.

A.3 Effect of accumulator injection upon the WBN Unit 2 steamline break analyses

The WBN Unit 2 steamline break analysis results show that the steamline break analysis without offsite power would yield a higher peak post-trip power level than would the steamline break analysis with offsite power. In the latter case, the peak post-trip power level is limited by the effect of borated water injection from the accumulators. The staff asked the applicant to address smaller steamline breaks with offsite power that would not depressurize to the accumulator injection setpoint, and consequently would not benefit from borated water addition from the accumulators.

The staff's question:

Support the claim that a minor secondary system pipe break would be less limiting than the major steam line rupture, since boric acid is supplied to the core by the accumulators during the major steam line rupture; but not necessarily during the minor secondary system pipe break.

The applicant's response:

Per the NRC approved topical report - WCAP-9226-P-A, "the largest double-ended steamline rupture at end of life, hot shutdown conditions with the most reactive rod cluster control assembly in the fully withdrawn position is a limiting and sufficiently conservative licensing basis to demonstrate that the Westinghouse PWR [pressurized-water reactor] is in compliance with 10 CFR 100 criteria for Condition II, III, and IV steamline break transients."

The response did not address the staff's concern, since all of the WCAP-9226 analyses were based upon the assumed delivery of 20,000 parts per million (ppm) boric acid to the core by the safety injection system. This addition of highly concentrated boric acid, early in the transient, when RCS pressure is significantly higher than the accumulator injection setpoint, would quickly render the core subcritical. The effect of boric acid from the accumulators would be relatively insignificant. The staff believes that the conclusion of WCAP-9226 was valid in 1978, when 20,000 ppm boric acid was available. It is not clear that this conclusion would still hold today, when 20,000 ppm boric acid is no longer available. This is why the staff made the distinction between steamline break sizes that cause accumulator injector and those that do not.

The staff also asked the following, related question:

What is the largest steamline break size that would not rapidly depressurize the RCS to the accumulator delivery setpoint (i.e., rapidly enough to make accumulator injection the primary mitigation function)?

The applicant's response:

All breaks assuming offsite power is available will depressurize to the accumulator injection pressure. The smaller (minor) breaks will reach the injection pressure later than the larger (major) breaks and smaller breaks result in less cooldown and a lower peak heat flux and are subsequently less limiting.

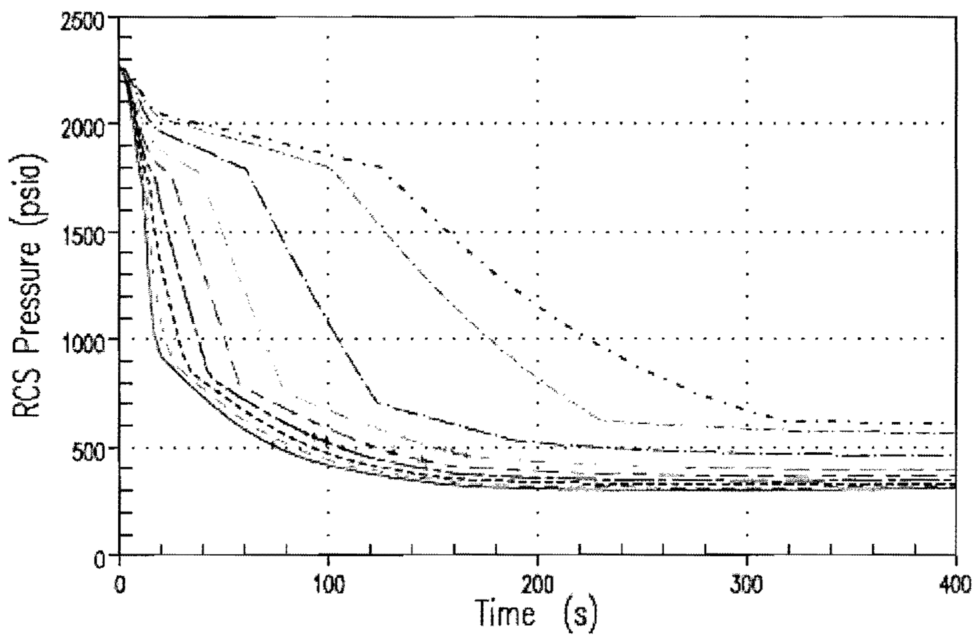
The response is repeated, when the staff asks this question again, in another context.

Provide a discussion to support the claim that the main steamline rupture is more limiting than the accidental depressurization of the main steam system given that boric acid is supplied to the core by the accumulators in the former case; but not in the latter case.

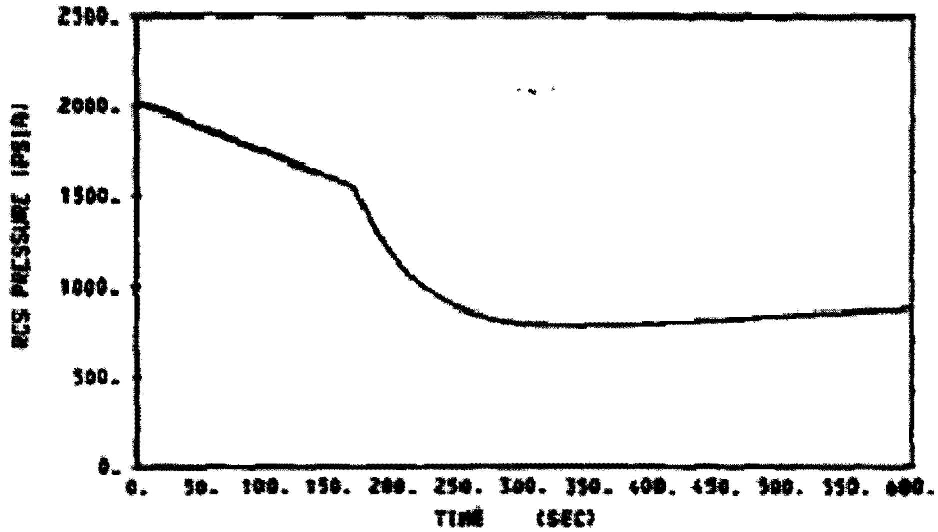
The applicant's response:

All breaks assuming offsite power is available will depressurize to the accumulator injection pressure. The accidental depressurizations will reach the injection pressure later than the main steamline rupture, and smaller breaks result in less cooldown and a lower peak heat flux and are subsequently less limiting.

The applicant supplemented its response with the results of a series of steamline break analyses, performed for various sizes, down to 0.05 ft^2 (top curve). In each case the accumulator injection setpoint is reached. The smaller sizes reach the accumulator injection setpoint later in the transient.



At the audit, the staff produced an RCS pressure transient representing a credible break (i.e., the spurious opening of a steam system valve with a throat area of 0.11 ft^2) that was copied from WCAP-12603, Point Beach's report for the reduction of safety injection system boron concentration (September 1990). The Point Beach transient was selected because, like the WBN Unit 2 transient, there is no heat addition (to keep pressure high) due to power generation. The break area is twice the break area of the top curve, and roughly equivalent to the break area of the curve just below the top curve. Since none of these three pressure transients stem from analyses that predict a return to criticality, there is no power generation to retard the depressurization transients. The WCAP-12603 transient (below) indicates that the accumulator setpoint is not reached.

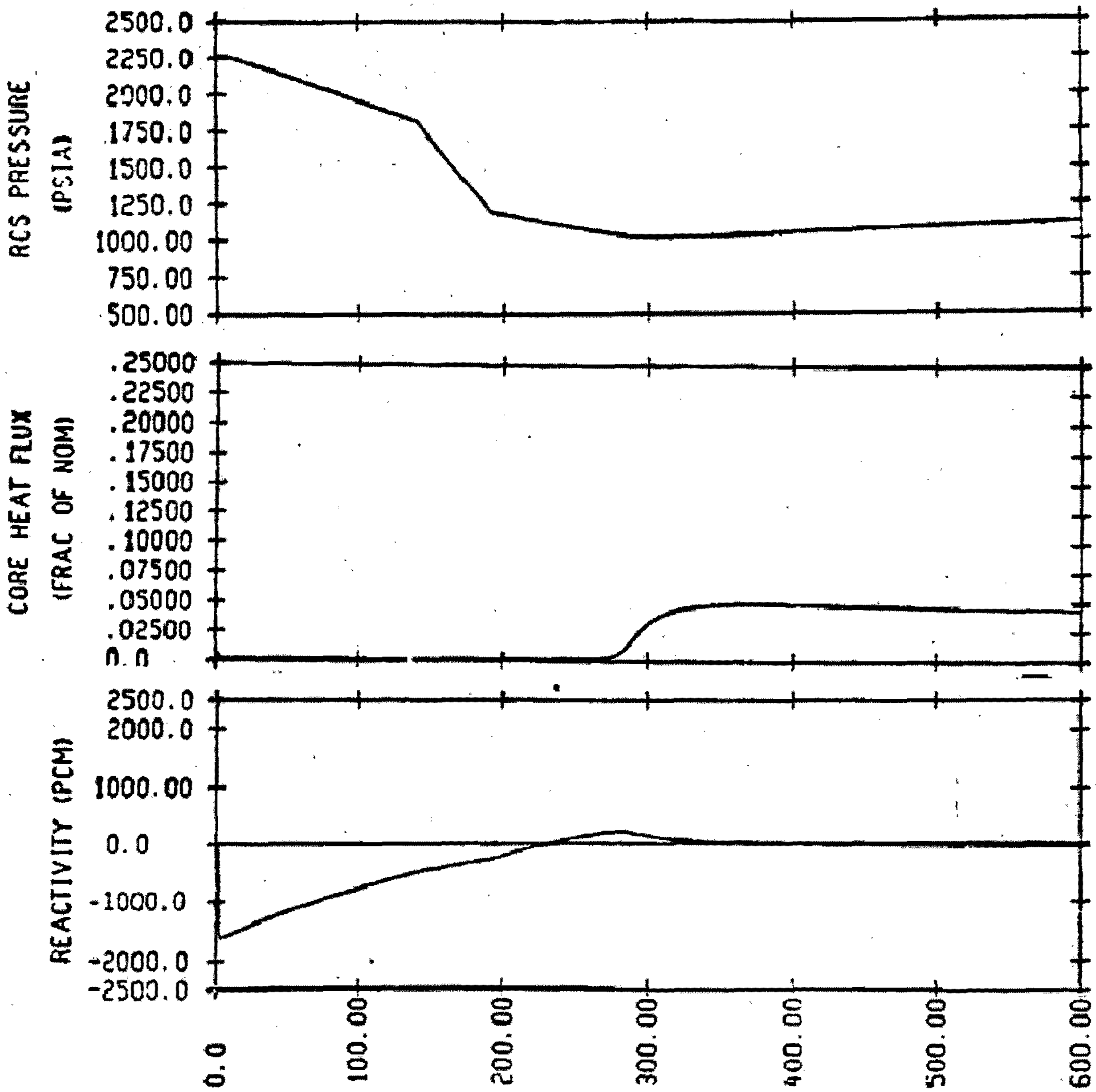


Indeed, it shows the beginning of a trend to pressurize the RCS, which, if allowed to continue, would eventually cause RCS pressure to reach the highest shutoff head of all the operating emergency core cooling system (ECCS) pumps or the opening setpoint of the pressurizer power-operated relief valves (PORVs), whichever is lower. In the case of WBN Unit 2, it would be the opening setpoint of the PORVs (see mass addition event analyses and methods, below). Since the RCS has been cooled by the blowdown of a steam generator, pressurized thermal shock could become a concern.

The corresponding steamline break analysis, for WBN Unit 1 (and therefore, WBN Unit 2), is found among the boron injection tank (BIT) concentration reduction case studies conveyed by letter dated May 2, 1984, from TVA to the NRC. The RCS pressure transient resulting from the credible break (0.11 ft^2), designated Figure 15.2-41 in the report, and depicted below, is not suitable for comparison to the prior cases, with respect to accumulator injection, since the analysis predicts a return to criticality, which leads to a sustained power generation of almost 5 percent of nominal. Consequently, this heat addition could tend to keep the minimum RCS pressure above the accumulator injection setpoint. As expected, the transient plot shows that the accumulator injection setpoint is not reached. However, it is interesting to note that, in 1984, this analysis predicted that the core would return to critical, whereas today, no return to criticality is predicted. In fact, no return to criticality is predicted for any break size up to and including 0.6 ft^2.

It was necessary to reach back decades to obtain the results of analyses of credible steamline break cases because Westinghouse no longer includes such analyses among the cases reported in FSARs or other licensing submittals. Today, they're systematically dismissed as being bounded by the major steamline rupture cases. WCAP-9226 examined a range of break sizes; but did not include the credible break size, 0.11 ft^2. It was not necessary to consider the credible break size because, in 1978 (when WCAP-9226 was issued), Westinghouse adhered to an internal acceptance criterion that did not permit the credible break to cause a return to criticality. Credible steamline break analyses consistently showed that 20,000 ppm boric acid, from the ECCS reached the core before it could reach criticality, and then the high concentration boric acid caused the reactivity to decrease rapidly. Since the core did not produce any power, it could not lead to DNB. It was always bounded by the major steamline break, by definition.

This is no longer true. In the years following the publication of WCAP-9226, Westinghouse has (1) reduced the concentration of boron in the BIT, and (2) abandoned its internal no-criticality acceptance criterion for credible steamline breaks. Today, credible steamline breaks can result in a post-trip return to criticality. Delivery of ECCS flow to the core no longer causes a steep drop in core reactivity. Instead, core reactivity falls to the critical level, and stays there until core boron concentration builds up to a level that would render the core subcritical, or until the positive reactivity insertion, due to the steamline break-induced cooldown, is ended (i.e., the faulted steam generator is empty, or nearly empty). The plots, below, which result from an analysis of the WBN Unit 1 credible steamline break, with reduced boron concentration in the BIT, are an example of this behavior. Since this resulting power generation could cause the DNBR safety limit to be violated during a credible steamline break, it can no longer be categorically stated that the major steamline break will always bound the credible break. There is the possibility that a small break, that does not cause the RCS to depressurize to the accumulator injection setpoint, could yield a lower DNBR than would a large steamline break, which receives the benefit of boron from the accumulators. This possibility is behind the staff's questions.



The same 1984 letter reports analyses of the 1.4 ft^2 steamline break with and without offsite power for WBN Unit 1. Here are the results, as compared to the current results:

1.4 ft^2 steamline break	WBN1 peak power (1984)	RCS flow	WBN2 peak power (2011)	RCS flow
with offsite power	~18% at 100 – 200 s	100%	1.6% at 57 s	100%
without offsite power	~14% at 100 – 200 s	< 10%	~ 7% at ~170 s	< 10%

Note that (1) the peak power level in the "limiting" case, with offsite power has dropped by more than a factor of ten, between 1984 and 2011, and (2) the "non-limiting" case, without offsite power, now produces a much higher power than the case with offsite power. The conclusion of WCAP-9226 was based, in part, on the premise that steamline break cases with offsite power always reached higher peak power levels than steamline break cases without offsite power.

B. Mass Addition Events

The applicant stated that analyses of the mass addition events: inadvertent operation of the ECCS at power, and chemical and volume control system (CVCS) malfunction, are currently in progress. The applicant plans to show that there is sufficient time for the operator to end the mass addition before the pressurizer fills. The applicant will submit the analyses when they're completed.

In the meantime, the applicant handed the staff a copy of TI-12.19, "Control of Time Critical Operator Actions", which includes a section for the inadvertent operation of the ECCS (Attachment 2), which specifies that operator actions to end the mass addition be completed within 10 minutes. The staff's comments regarding this document are:

- There is no corresponding attachment for the CVCS malfunction event.

- In Attachment 2, Section1.0, "Action", it states, "Terminate injection flow at 10 minutes via operator action in order to prevent pressurizer pressure from reaching the opening pressure of the safety valves due to continued inventory addition." The staff disagrees with this statement. Operator action should be completed within the time period that the inadvertent operation of the ECCS analysis indicates is acceptable, not at 10 minutes. The 10 minute value is often interpreted as the minimum acceptable reference period. The purpose is not to prevent pressurizer pressure from reaching the opening pressure of the safety valves. It is to prevent the pressurizer from filling. Opening the pressurizer PORVs and/or the safety valves, and relieving steam, is acceptable.

- The acceptance criterion in Section 2.2D is no longer in effect (see NUREG-0800, Revision 3).

- The Westinghouse acceptance criterion in Section 2.3A is not acceptable to the NRC. In fact, this is the reason the applicant is performing a new analysis for the inadvertent operation of the ECCS event.

- The Westinghouse acceptance criterion in Section 2.3B misses the point. The DNBR concern is trivial; but not for the reason stated. Since this event is considered with respect to the effects of mass addition (not reactivity or power), the inadvertent operation of the ECCS analysis is performed assuming that the ECCS boron concentration is the same as the core boron concentration. Since the ECCS actuation signal also generates a reactor trip signal, there is no power generation at any time during the event, and therefore, no concern regarding the potential for DNB.

- The Westinghouse acceptance criterion in Section 2.3C is not understood. There is no definition of the "expansion rate of the RCS pressure." There is also no mention of whether the safety valves are relieving water or steam.

- The two references, in Section 3.0, to Westinghouse Nuclear Safety Advisory Letter (NSAL)-93-013, "Inadvertent ECCS Actuation at Power" are not relevant. The new analyses need not apply the guidance in either of those documents.

The staff had documented its objections to the NSAL-93-013 strategy in RIS 2005-029. For example, the staff disagrees with the logic that is expressed in the following paragraph from NSAL-93-013 (and NSAL-00-013 and NSAL-07-10):

Without appropriate operator action to terminate safety injection flow prior to reaching a water-solid pressurizer condition, the Inadvertent ECCS Actuation at Power event may progress from a Condition II to a more severe Condition III LOCA event as described above. While this occurrence may result in a violation of one of the applicable licensing basis criterion for a Condition II event, it is not considered a significant safety concern. As a LOCA event, discharge of coolant out of the pressurizer safety relief valves and PORVs due to ECCS flow is not significantly adverse relative to other Condition III LOCA events currently analyzed. This is because the pressurizer is located on the hot leg (a hot leg LOCA being less severe than a cold leg LOCA) and because the Inadvertent ECCS Actuation at Power event typically models maximum ECCS flow (to maximize the effects of the initiating event), which is a benefit for LOCA. As such, the Inadvertent ECCS Actuation at Power induced LOCA is bounded by the existing small break LOCA analyses.

Westinghouse claims that violation of the acceptance criterion that prohibits a Condition II event from developing into a more severe Condition III event does not pose a significant safety concern, since the resulting LOCA would be no worse than currently analyzed LOCAs. The NRC staff observes that LOCAs are Condition III events (or Condition IV events, depending on break size); evaluated according to Condition III acceptance criteria. The LOCA that is initiated by the inadvertent ECCS operation event has a Condition II frequency of occurrence, and ought to be evaluated according to Condition II acceptance criteria. In other words, this is a LOCA that must not cause any fuel damage. Westinghouse's assessment, which is based upon an apples-to-oranges comparison, undermines the acceptance criterion that prohibits the development of a more severe event, in the less frequent event class, initiated from a more frequent, less severe event.

Westinghouse stated that it was no longer recommending the use of this approach. The applicant is asked to supply a copy of Westinghouse Letter LTR-A-09-181, Revision 1, dated November 17, 2009.

C. RCS and main steam system overpressure analysis

The applicant had provided the results of WBN Unit 2 overpressure analyses that were performed according to the guidance of Standard Review Plan 5.2.2, Overpressurization Protection. The results indicated that the overpressure safety limits are met. The applicant was asked by the staff to provide the full results, including the event description, analysis methods and assumptions, and transient plots.

D. Loss-of-Coolant Accident

D.1 LOCA and post-LOCA long term cooling

As required by 10 CFR 50.46, each pressurized light water nuclear power reactor must be provided with an ECCS that must be designed so that its calculated cooling performance following postulated LOCAs conforms to, among others, a requirement for long-term cooling. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

To compensate for positive reactivity effects, the PWR ECCS contains borated water, with a boron concentration maintained at a level that is governed by the facility TSs. Due to the cooling phenomena present in a reactor core following a LOCA, boron tends to concentrate in the core region. In extreme cases, the concentration of boron could become sufficiently high as to begin precipitating in the core. This precipitation could inhibit decay heat removal and cause unacceptable temperature increases over a period of time following the LOCA.

During the audit meeting at the Westinghouse offices in Rockville on March 15, 2011, the licensee provided data and information pertaining to RAIs on boric acid precipitation that included:

1. Geometric data for the vessel
2. Loop resistance data
3. Sump transient boric acid concentration vs. time
4. A copy of the Westinghouse boric acid precipitation analysis
5. Top and bottom skewed power shapes

Transient plots of the following parameters for the large break LOCA with downcomer boiling transient were also requested, including:

1. Downcomer Level and fluid temperatures
2. Containment pressure
3. RCS pressure
4. Heat transfer coefficient at the hot spot
5. Peak centerline temperature (PCT)
6. Sink Temperature at the PCT location
7. Core level
8. Low pressure safety injection and high pressure safety injection mass flow rates
9. Was heat transfer from the barrel wall and thermal shield also included in the evaluation?
10. Refueling water storage tank (RWST) temperature

It was also stated that the RWST does not drain prior to the PCT for the downcomer boiling case. An explanation for how condensation was treated for this event was requested and whether condensation maximized? The effect of time step on the results was also requested and whether this included time steps down to and including 0.001 second. Confirmation of the correct conductivity in the downcomer walls that was used in the evaluation was also requested as well as the values for the lateral k-factors in the downcomer (were they based on Idlechik?).

In regard to the limiting large break LOCA transient, the decay heat multiplier used in the transient for this 95/95 case was also requested.

A nodalization diagram was also requested for the WCOBR/TRAC model.

The staff also requested an analysis of a bottom mounted instrument tube failure. During the audit a generic report on instrument tube failure was provided. The staff again asked for a specific analysis of a tube failure for WBN.

April 27, 2011

Mr. Ashok S. Bhatnagar
Senior Vice President
Nuclear Generation Development
and Construction
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 – AUDIT REPORT OF WESTINGHOUSE DOCUMENTS RELATING TO FINAL SAFETY ANALYSIS REPORT ACCIDENT ANALYSES (TAC NO. ME4620)

Dear Mr. Bhatnagar:

The U.S. Nuclear Regulatory Commission (NRC) staff performed an audit of Westinghouse documentation that supports Tennessee Valley Authority's license application for Watts Bar Nuclear Plant, Unit 2. The audit took place on March 15, 2011, at the Westinghouse office located in Rockville, Maryland. Enclosed is the audit summary report prepared by the NRC staff.

If you should have any questions, please contact me at 301-415-2048.

Sincerely,

/RA/

Justin C. Poole, Project Manager
Watts Bar Special Projects Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-391

Enclosure:
Audit Summary

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NAME	JPoole	BClayton	AUlses*	SCampbell
DATE	4/27/11	4/25/11	4/12/11	4/27/11

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