ArevaEPRDCPEm Resource

From: WELLS Russell D (AREVA NP INC) [Russell.Wells@areva.com]
Sent: Wednesday, November 25, 2009 10:27 AM
To: Tesfaye, Getachew
Cc: Pederson Ronda M (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 241, FSAR Ch 15, Supplement 1
Attachments: RAI 241 Supplement 1 Response US EPR DC.pdf

Getachew,

AREVA NP Inc. (AREVA NP) provided a response to 1 of the 6 questions of RAI No. 241 on August 5, 2009. The attached file, "RAI 241 Supplement 1 Response US EPR DC.PDF" provides technically correct responses to the remaining 5 questions, as committed.

The following table indicates the respective pages in the response document, "RAI 241 Supplement 1 Response US EPR DC.PDF," that contain AREVA NP's response to the subject questions.

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This concludes the formal AREVA NP response to RAI 241, and there are no questions from this RAI for which AREVA NP has not provided responses.

Sincerely,

(Russ Wells on behalf of)
Ronda Pederson
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Licensing Manager, U.S. EPR Design Certification
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From: Pederson Ronda M (AREVA NP INC)
Sent: Wednesday, August 05, 2009 2:48 PM
To: Tesfaye, Getachew
Cc: BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); GUCWA Len T (EXT)
Subject: Response to U.S. EPR Design Certification Application RAI No. 241 (2769, 2804),FSAR Ch. 15

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 241 Response US EPR DC.pdf" provides a technically correct and complete response to 1 of the 6 questions.

The following table indicates the respective pages in the response document, "RAI 241 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 241 — 15.02.01-15.02.05-9	2	3
RAI 241 — 15.06.05-51	4	4
RAI 241 — 15.06.05-52	5	5
RAI 241 — 15.06.05-53	6	6
RAI 241 — 15.06.05-54	7	7
RAI 241 — 15.06.05-55	8	8

A complete answer is not provided for 5 of the 6 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 241 — 15.06.05-51	December 3, 2009
RAI 241 — 15.06.05-52	December 3, 2009
RAI 241 — 15.06.05-53	December 3, 2009
RAI 241 — 15.06.05-54	December 3, 2009
RAI 241 — 15.06.05-55	December 3, 2009

Sincerely,

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From: Tesfaye, Getachew [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Tuesday, July 07, 2009 7:57 AM
To: ZZ-DL-A-USEPR-DL
Cc: Liang, Chu-Yu; Forsaty, Fred; Lu, Shanlai; Donoghue, Joseph; Carneal, Jason; Colaccino, Joseph; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 241 (2769, 2804),FSAR Ch. 15

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on June 5, 2009, and discussed with your staff on July 2, 2009. Draft RAI Question 15.06.05-50 was deleted as a result of that discussion. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 984

Mail Envelope Properties (1F1CC1BBDC66B842A46CAC03D6B1CD41023F17C9)

Subject: Response to U.S. EPR Design Certification Application RAI No. 241, FSAR Ch 15, Supplement 1
Sent Date: 11/25/2009 10:26:54 AM
Received Date: 11/25/2009 10:27:12 AM
From: WELLS Russell D (AREVA NP INC)

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Files	**Size**	**Date & Time**
MESSAGE	4091	11/25/2009 10:27:12 AM
RAI 241 Supplement 1 Response US EPR DC.pdf		3080660

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

Response to

Request for Additional Information No. 241, Supplement 1

7/07/2009

U.S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 15.02.01-15.02.05 - Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)
SRP Section: 15.06.05 - Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary
Application FSAR Ch 15

QUESTIONS for Reactor System, Nuclear Performance and Code Review (SRSB)

Question 15.06.05-51:

If the two-phase mixture level drops below the TAF anytime during the U.S. EPR long term cooling phase of a LOCA, cladding heatup and oxidation can result.

Please provide the results of a thermal-hydraulic analysis quantifying the two-phase mixture level within the reactor core barrel during the long term cooling of the U.S. EPR core under the most limiting break size, break location and ECCS performance conditions. Discuss the conservatism of the obtained results. For important modeling parameters that are expected to vary within a certain range, substantiate the conservatism in selecting the value for each parameter or provide sensitivity assessments over the expected range of variation.

This question is as a follow-up to the reactor systems audit held on April 21-24, 2009.

Response to Question 15.06.05-51:

A quasi-steady static-balance analysis approach is used to calculate the core two-phase mixture level as a function of decay power during the post-reflood period. The ANS 1973 Standard is used to relate decay power to time after reactor shutdown. Low head safety injection (LHSI) remains operable and maintains postulated water elevation. The static balance approach shows that the minimum core collapsed liquid level is the same as the elevation at the top of the cross-over pipe U-Bend (loop seal). The two-phase mixture level rises above the collapsed liquid level because of boiling in the core. Computations, using three different two-phase flow correlations, show that the top of the active fuel (TAF) remains covered by a two-phase mixture for at least 1157 days into the post-reflood period using nominal input parameters. Using other than nominal input parameters does not change that result. Thus, fuel rod temperatures remain near the saturation temperature and fuel rod rods remain cooled and without added cladding oxidation in the post-reflood period. The elevation at the top of the cross-over piping U-bend (loop seal) is only 0.1 ft below the TAF and that is important to this result.

EGG-TFM-7993 (Reference 1) presents a long term core cooling assessment in the post-reflood period by using a quasi-steady static-balance approach and RELAP5. Reference 1 concluded that the simplified analysis was conservative relative to a full system analysis using RELAP5.

Figure 15.06.05-51-1 shows the concept and essential features of the quasi-steady static-balance model. The assumptions and description are as follows:

1) The system is assumed to have reached a quiescent state without significant fluid dynamic effects.

2) Elevations are relative to the bottom of the active fuel. The TAF is at elevation Z_{CORE}. The top of the cross-over pipe U-bend (loop seal) is at elevation Z_{LS}. Water on the steam generator side of the loop seal (i.e., Z_3) is shown at an elevation below that of the loop seal (i.e., Z_{LS}). The modeling assumes that the depression below the loop seal is null, so, Z_3 ranges from Z_{LS} to Z_{CL}.

3) LHSI is operational and injects sufficient water to offset the loss through the largest break and maintains postulated (conservative) water levels shown in Figure 15.06.05-51-1. LHSI capacity for a single pump at 14.5 psia is 334 lb/second, which provides makeup water to the core where the steaming rate is less than 10 lb/second in the post-

reflood period of interest. With adequate water inventory in the in-containment refueling water storage tank (IRWST), LHSI injection continuously delivers water to the primary system, as necessary.

4) When the LHSI system is switched to hot leg injection, it will deliver more than 75 percent of its subcooled water to the hot leg where it can condense steam and also deliver a large quantity of water directly to the top of the core; however, hot leg injection is not credited in this analysis, which is a conservative assumption.

5) A cold leg break between the reactor coolant pump and reactor vessel is the most limiting location because of restricted steam venting though the loop seal as shown in Figure 15.06.05-51-1. Other break locations do not impose significant limitations to the venting of steam and do not cause a depression of the collapsed liquid level in the core. The analysis is independent of break size and geometry because of the conservative assumption that LHSI maintains postulated water levels.

6) Decay heat in the core generates steam. Subcooled LHSI water enters the downcomer and flows to the core to replenish the water that has evaporated to steam.

7) Steam generators are cooled down and are neither a heat source, nor a heat sink for primary side steam. Reflux condensation of steam in the steam generators is excluded from this analysis because of its ability to return water to the upper plenum, providing long term core cooling.

8) There are two escape flow paths for steam generated by decay heat in the core: Path 1 and Path 2. Path 1 is through the downcomer-to-hot-leg bypass and through the downcomer-to-upper-head bypass; however, the steam flow path is in the reverse direction to the downcomer, to the cold leg and to the break. Path 2 allows steam to flow through the hot leg, steam generator, cross-over pipe (loop seal), and reactor coolant pump to the cold leg, then to the break.

9) The cold leg cross-over pipe may hinder steam ventilation. The U-bend in the cross-over pipe can pass steam only if the water elevation (i.e., Z_3) is sufficiently depressed on the steam generator side. The elevation at the top of the U-bend, Z_{LS}, is 0.1 ft below the elevation at the TAF, Z_{CORE}.

10) Pressure drops are defined by elevation head differences in the loops and reactor vessel. Fluid frictional resistance is very small for Paths 2 and 3 and hydrostatic head dominates for the post-reflood period. Pressure drop in the bypass is defined by flow resistance modeling with loss coefficients derived from a separate analysis of the U.S. EPR reactor vessel. The static-balance equations show that the collapsed liquid level, Z_1, is equal to the water elevation on the steam generator side of the loop seal, Z_3.

11) The water and steam are saturated at the selected primary system pressure except for the water in the downcomer and core inlet. The subcooled water temperature at the core inlet is equivalent to the LHSI temperature. Subcooled water at the core inlet produces a lower two-phase mixture level than if saturated water is assumed.

12) The core is assumed to have a uniform axial and radial power distribution as indicated in Reference 1 and “The Prediction of Two-Phase Mixture Level and Hydrodynamically-

Controlled Dryout under Low Flow Conditions" (Reference 2). This assumption is sufficient for this long term cooling assessment and allows a closed form analytical solution for the two-phase mixture level, Z_2, in terms of the core power and collapsed liquid level, Z_1, and the start of boiling at elevation Z_0.

13) The loops are assumed to be symmetric regarding loop seal filling and venting.

Static-balance modeling is used to define the collapsed liquid level in the core and the associated water levels. There are three periods of interest:

1) High decay power in the post-reflood period causes the steam generated within the active core to vent through Path 1 and Path 2 as shown in Figure 15.06.05-51-1. The water elevation Z_3 is depressed sufficiently to allow steam to pass through the loop seal. The two-phase mixture in the vertical leg on the pump side of the loop seal reduces the hydraulic head and the core collapsed liquid level, Z_1, is near the elevation of the cold leg. Thus, the collapsed liquid level is well above the TAF. The two-phase mixture level, Z_2, is also well above the TAF.

2) As the power decreases, the steam generation rate decreases. A power transition is reached where all steam is vented through Path 1 (bypass) and no steam is vented through Path 2 (loop seal). At this transition power, the core collapsed liquid level, Z_1, is at its minimum and is equal to the loop seal elevation, Z_{LS}.

3) Further decrease of power reduces the steam generation rate. A reduction of steam generation rate reduces the steam pressure at the steam generator side of the loop seal, and the water elevation Z_3 increases toward the elevation of the cold leg. The core collapsed liquid level also rises because $Z_1 = Z_3$.

Energy balance for fluid in the core defines the start of bulk boiling and the steam generation rate. Subcooled water enters the bottom of the core and is heated to the start of bulk boiling at elevation Z_0. Boiling in the two-phase region generates steam that produces a two-phase mixture level, Z_2, which is above the collapsed liquid level, Z_1.

The static-balances define the elevation of collapsed liquid level, Z_1, and the energy balance defines the elevation at the start of boiling, Z_0. The two-phase mixture elevation, Z_2, is defined by using the three void-fraction correlations applied in Reference 1. By using a uniform power distribution in the core, it is possible to develop a closed form analytical solution for the two-phase mixture level (i.e., Z_2) in terms of the collapsed liquid level (i.e., Z_1), start of boiling elevation (i.e., Z_0), and decay power. The closed form solution for the Zuber-Findlay correlation is shown in Reference 2 and solutions for the Wilson and Cunningham-Yeh correlations are shown in Reference 1. The closed form solutions for this analysis are identical to those in Reference 1 except for subcooling at the core inlet that defines Z_0. The inclusion of subcooled water produces a lower two-phase mixture level, Z_2, and its inclusion is conservative for this analysis. While the solution for the mixture level Z_2 applies to the active core, a solution greater than Z_{CORE} implies that the mixture level is above the TAF.

The three void fraction correlations are compared with experimental data in Reference 2. Those comparisons suggest that the Zuber-Findlay correlation may over-predict Z_2 and the Cunningham-Yeh correlation may under-predict Z_2. The Wilson correlation appears to be the closest to experimental data for the related application in Reference 2. All three correlations are included to show the range of variation.

Table 15.06.05-51-2 defines the input selections for a nominal case and offsets for selected parameters of interest.

Table 15.06.05-51-3 presents a summary of results. The minimum collapsed liquid, Z_1, is 13.68 ft, which is 0.1 ft below the top of the active core. The time it takes to reach the minimum level depends on parameters that affect the flow split between the bypass and the loops. Pressure and bypass flow resistance are two parameters of interest. Using nominal bypass flow resistance and atmospheric pressure for Case 1, the minimum collapsed liquid level occurs beyond the 1157 days of the computation. Increasing the pressure in Case 2 increases the steam density and bypass flow rate. The minimum collapsed liquid level, Z_1, is reached in 604 days. Decreasing the bypass flow resistance coefficient in Case 3 increases the bypass flow further and the minimum is reached in 146 days. While there is variability in the bypass flow resistance, the impact of the variability affects only the time necessary to reach the minimum collapsed level.

In all cases, and for all three two-phase flow correlations, the two-phase mixture level, Z_2, is always above the TAF. The Cunningham-Yeh correlation produces the lowest two-phase levels; thus, it is the most conservative.

Figure 15.06.05-51-2 shows the results of the nominal Case 1. The core collapsed liquid level, Z_1, is initially near the elevation of the cold leg and then decreases with power. However, there is no minimum for Z_1 during the time of this computation and Z_1 remains above the TAF. The minimum occurs beyond 1157 days. The bypass is highly restrictive; therefore, steam is vented through both the loops and bypass for this computation. The two-phase mixture levels, Z_2, are above the TAF for all three two-phase modeling approaches. The Cunningham-Yeh correlation produces the most conservative results consistent with Reference 1. The fuel rods are always covered by two-phase coolant flow.

Figure 15.06.05-51-3 shows a plot of the two-phase mixture elevations at the increased pressure of 32 psia. The increased pressure increases the density of the steam and the steam bypass flow rate, which creates a minimum collapsed liquid level, Z_1, that is reached in 604 days and reduces the two-phase mixture levels from those in nominal Case 1. The minimum Z_1 is at the top of the cross-over pipe U-bend (loop seal) and 0.1 ft below the TAF. The power at minimum Z_1 is the transition power where the loop steam flow stops because the loop seal blocks steam flow causing it to vent through the bypass. As the core power decreases below the transition power, Z_1 increases toward the elevation of the cold leg as a result of the water level increase, Z_3, on the steam generator side of the loop seal. The two-phase mixture levels, Z_2, are above the TAF using all three modeling approaches. The fuel rods are always covered by two-phase flow.

Figure 15.06.05-51-4 shows a plot of the two-phase mixture elevations when the bypass flow resistance is decreased from the nominal resistance to the minimum resistance. The collapsed liquid level is initially near the elevation of the cold leg, and then decreases to the minimum value within 147 days. The overall result is similar to Case 2, but with earlier timing of minimum Z_1. The minimum Z_1 is again at the elevation of the loop seal and is 0.1 ft below the TAF. The two-phase mixture levels, Z_2, are above the TAF using all three modeling approaches. The fuel rods are always covered by two-phase flow.

Table 15.06.05-51-4 shows a list of parameters that affect the modeling used for this analysis. The design elevations are of high importance to the analysis. In particular, the top of the cross-over pipe U-bend (loop seal) is only 0.1 ft below the TAF.

Bypass flow resistance is uncertain and affects the timing of the minimum collapsed liquid level, Z_1, in the core. The analysis shows that, while the bypass flow resistance can affect timing, the minimum Z_1 is always at the loop seal elevation, Z_{LS}.

The two phase flow correlations are an essential part of the analysis; they show that the fuel is always covered by a two-phase mixture.

The quasi-steady static-balance modeling approach, with a range of input choices, shows that the U.S. EPR core remains covered with a two-phase mixture during the post-reflood period. Fuel rod heatup is not credible.

References

1. EGG-TFM-7993, “Long Term Recovery of Westinghouse Pressurized Water Reactors Following a Large Break Loss of Coolant Accident,” C. D. Fletcher and R. A. Callow, Idaho National Engineering Laboratory, EG&G Idaho, Inc., Idaho Falls, ID, February 1988.

2. “The Prediction of Two-Phase Mixture Level and Hydrodynamically-Controlled Dryout under Low Flow Conditions,” K. H. Sun, R. B. Duffey and C. M. Peng, Int. J. Multiphase Flow, Vol. 7, No. 5, pp. 521-543, 1981.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Table 15.06.05-51-1—Elevations Relative to Bottom of the Active Fuel

Location	Elevation from BAF, ft
Bottom of the Active Fuel (BAF)	0
Top of the Active Fuel, Z_{CORE}	13.78
Cross-over Pipe, top, loop seal, Z_{LS}	13.68
Cold Leg (center), Z_{CL}	20.87
Upper Plenum (same as cold leg), Z_{UP}	20.87

Table 15.06.05-51-2—Computational Case Matrix

	Nominal (Case 1)	Offsets from Nominal
Pressure, psia	14.7	32.0 (Case 2)
Bypass Flow Resistance, $1/ft^4$	303.24 (nominal flow rate)	50.33 (maximum flow rate) (Case 3)

Table 15.06.05-51-3—Summary of Results

	Pressure	Bypass, K/A^2	$Z_{1, MIN}$	Time of $Z_{1,MIN}$	Z_2
Case 1	14.7 psia	303.24 ft^{-4}	13.68 ft not reached in 1157 days	> 1157 days	> Z_{CORE}
Case 2	32.0 psia	303.24 ft^{-4}	13.68 ft	603.0 days	> Z_{CORE}
Case 3	14.7 psia	50.33 ft^{-4}	13.68 ft	145.4 days	> Z_{CORE}

Table 15.06.05-51-4—List of Model Parameters

Item	Importance
Design Elevations	High - The relative elevation of the loop seal and top of the active fuel is very important to the analysis. The U.S. EPR loop seal elevation is only 0.1 ft below the top of the active core.
Bypass Flow Resistance	High - Affects steam flow split and timing
Two-Phase Flow Correlations	High - Correlations show a range of results and all produce a two-phase mixture level that covers the fuel.
LHSI Operation	High - The operation of LHSI is fundamental to the assumptions of this analysis.
Decay Heat Table	Moderate - Decay heat table places the solution in time.
Pressure	Moderate - Affects steam flow split and timing. Near atmospheric pressure late in post-reflood period.
Subcooled Core Inlet	Moderate - Subcooling reduces the two-phase mixture level.
Elevation of Water in Primary Piping	Low - Affects the hydrostatic head before and after the transition power. Minor affect on flow split.
Void Fraction in Vertical Pipe on Pump Side of Loop Seal	Low - Affects the hydrostatic head before the transition power. Minor affect on flow split.

Figure 15.06.05-51-1—Modeled Steam Flow Paths and Elevations

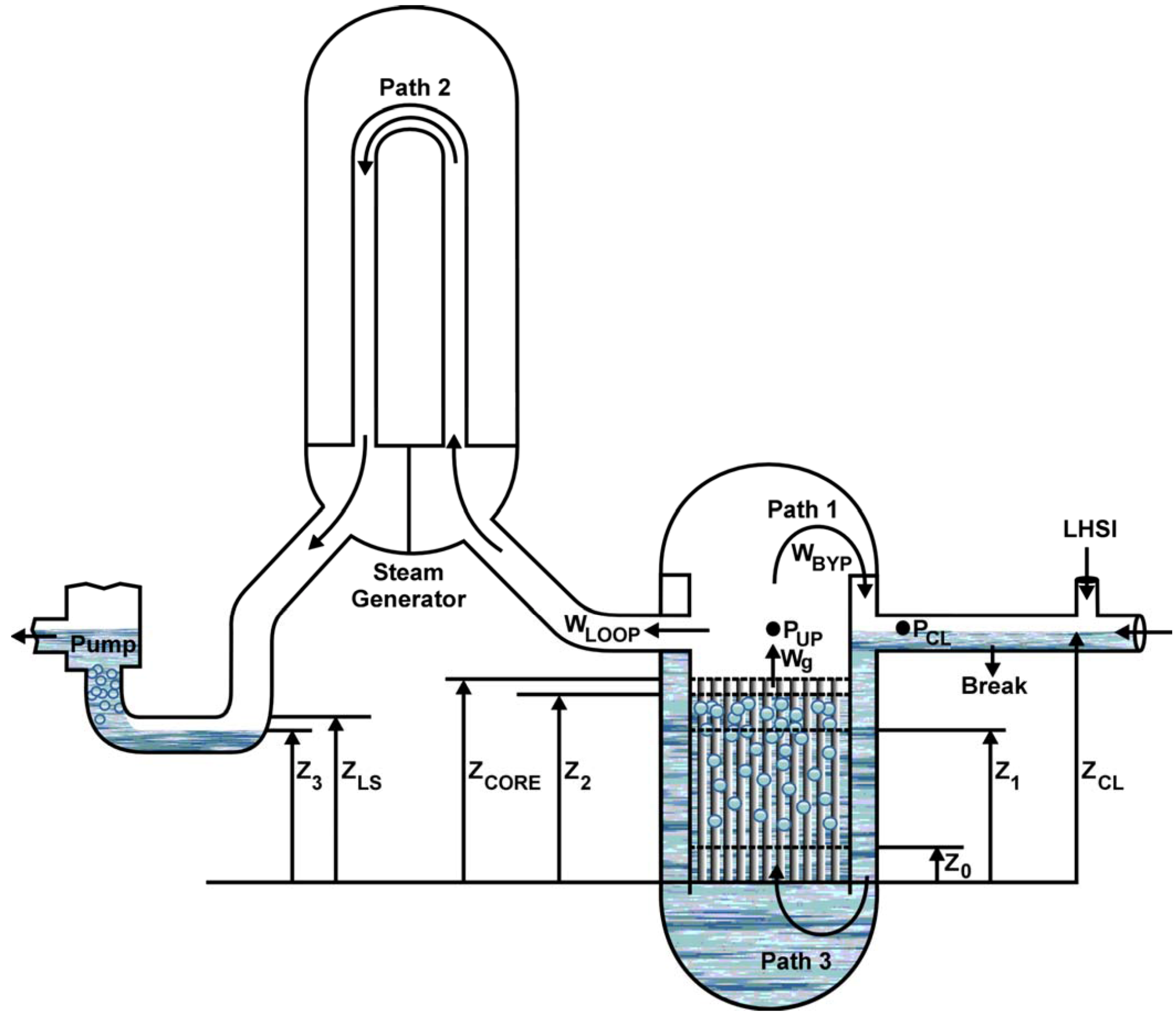


Figure 15.06.05-51-2—Coolant Elevations: P = 14.7 psia, K/A^2 = 303.24 ft^{-4}

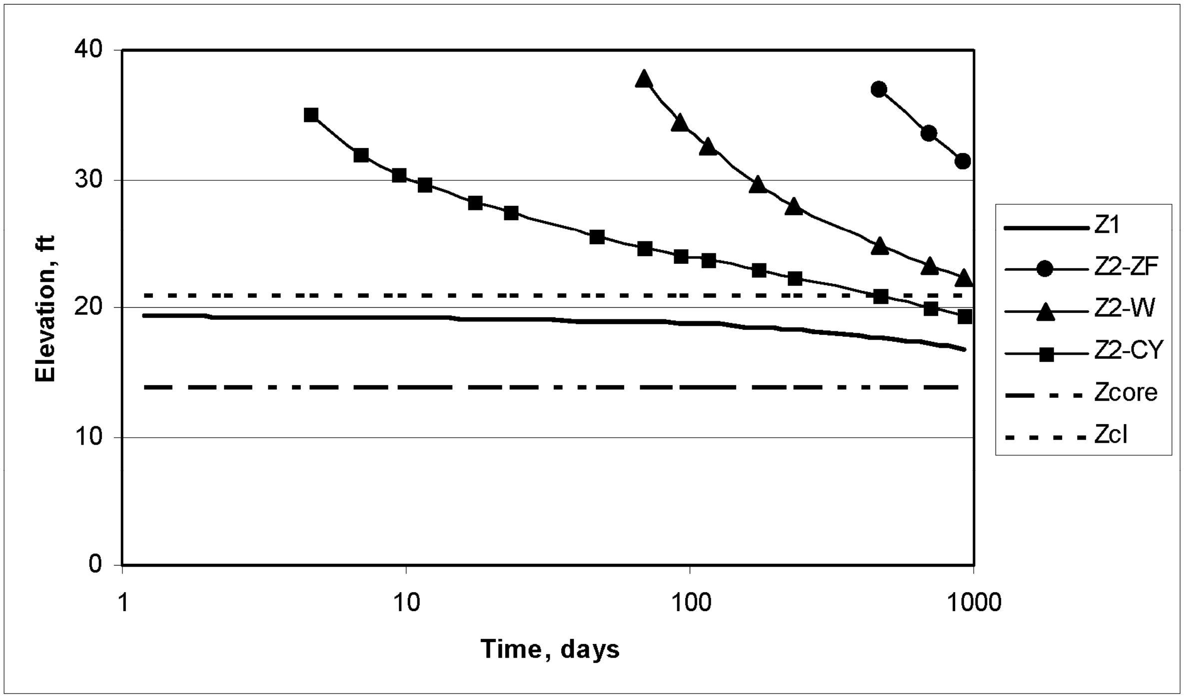


Figure 15.06.05-51-3—Coolant Elevations: P = 32.2 psia, K/A^2 = 303.24 ft^{-4}

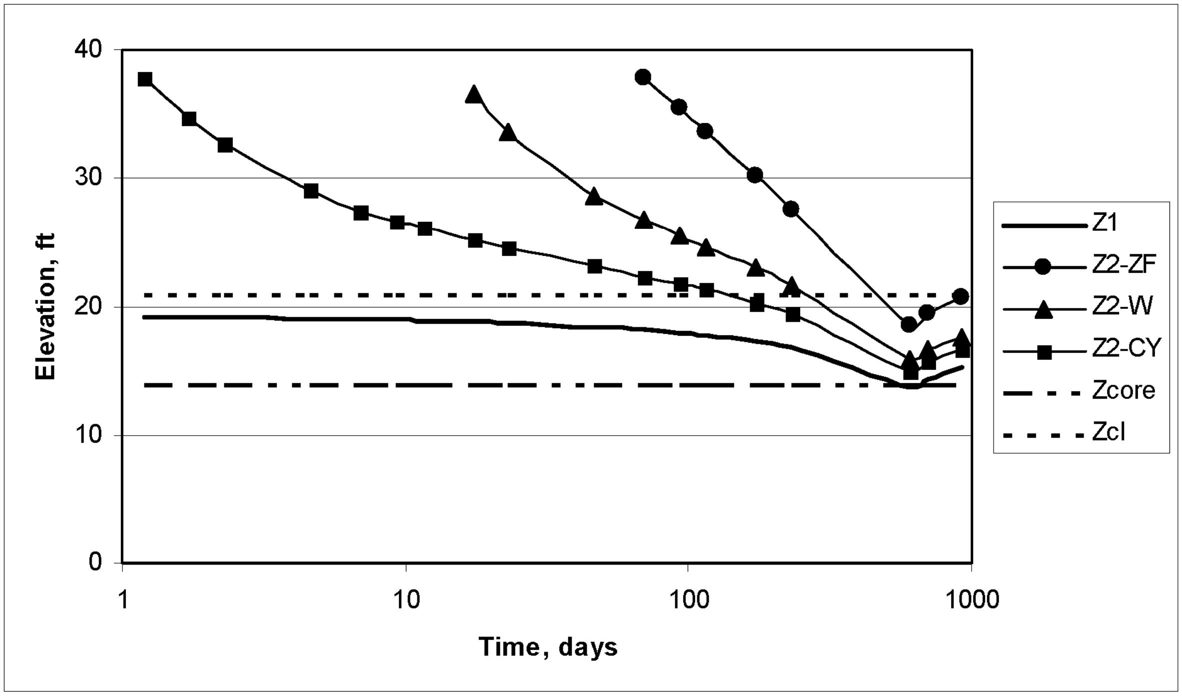
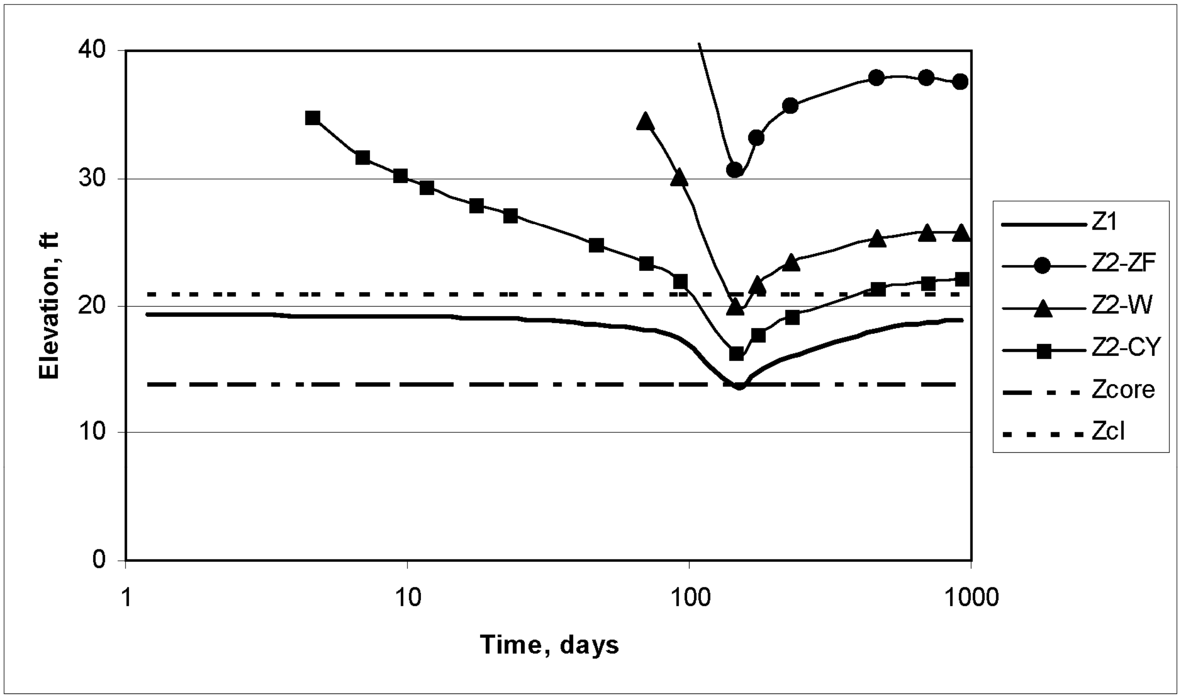


Figure 15.06.05-51-4—Coolant Elevations: P = 14.7 psia, K/A^2 = 50.33 ft^{-4}



Question 15.06.05-52:

In the mass balance approach for computing boric acid accumulation due to steaming in the U.S. EPR core during post-LOCA cooling, the licensee assumed a certain control (mixing) volume consisting of five different regions (sub-volumes) with individually assigned void fractions. This resulted in assumed core-region liquid volume fractions of 60% and 58% of the total liquid inventory within the mixing volume for the LBLOCA and SBLOCA analyses, respectively.

As the predicted precipitation time is proportional to the assumed liquid content within the mixing volume chosen, please demonstrate that the void fraction values, as applied to the individual mixing sub-regions, are conservative under the thermal hydraulic conditions encountered in the sub-regions over the period of interest. In particular, substantiate the applicability of a single void fraction value for computing the liquid amount within an entire sub-volume and explain any averaging assumptions with respect to both space and time.

This question is a follow-up to the reactor systems audit held on April 21-24, 2009.

Response to Question 15.06.05-52:

In the boron precipitation analysis, the large break loss-of-coolant accident (LBLOCA) is more limiting than small break LOCA break loss-of-coolant accident (SBLOCA) because the liquid volumes in the core regions, and thus the concentrating volume, are smaller. The LBLOCA concentrating volume was taken as the volume within the five sub-volumes at the end of peak cladding temperature (PCT) analysis. These five sub-volumes (i.e., heated core region, heavy reflector, guide tube regions, lower plenum, and upper plenum up to the hot legs) are split into more nodes within the S-RELAP5 model. The liquid contents in the five sub-volumes were calculated using all of the nodal void fractions in each particular sub-volume. This resulted in a total volume of 617 ft^3. The void fractions are representative void fractions for each of the five sub-volumes at the end of the PCT transient. For example, in the realistic LBLOCA (RLBLOCA) model there are 27 axial nodes and 4 radial regions in the core. The volume of liquid and the representative void fraction were calculated as:

Core Region $$\mathrm{V}_{\mathrm{liquid}} = \sum_{n=1}^{108} (1 - VF_n) \cdot \mathrm{V}_{\mathrm{total,\,n}}$$

Representative $$VF_{\mathrm{Core\ Region}} = 1 - \frac{\mathrm{V}_{\mathrm{liquid\ \ in\ core\ region}}}{\mathrm{V}_{\mathrm{total\ \ core\ region}}}$$

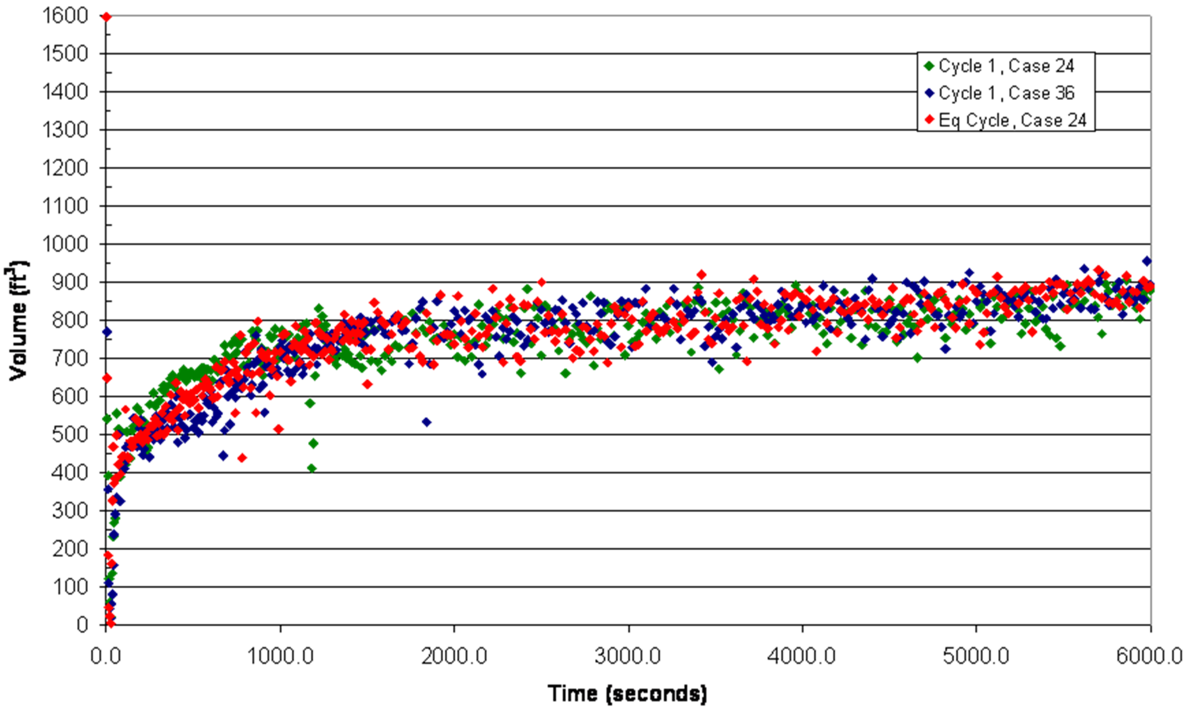
Using liquid volumes at the end of the PCT transient was assumed to be appropriately conservative for evaluating boron buildup in the core region to prevent boron precipitation. The lower plenum was included in the concentrating volume because of the recirculation pattern which develops from the core exit region through the heavy reflector, the peripheral core assemblies, and the guide tubes back into the lower plenum (See the Response to RAI 15.06.05-55).

To confirm that the model was conservative the volumes at the end of the PCT transient for all 118 cases (59 cases for cycle 1 and 59 cases for the equilibrium cycle) of the RLBLOCA analysis have been examined. Three cases that contained the least volume in the concentrating region were re-run with the transient analysis extended to 6000 seconds. The recirculation pattern through the five sub-volumes developed with the downflow rates exceeding the evaporation rate at 1000 seconds into the transient. As shown in Figure 15.06.05-52-1, the three cases show that the volume in the five sub-volume concentration volume exceeds the 617 ft^3 used in the original analysis with 750 ft^3 to 900 ft^3 at 6000 seconds. Although all three of these cases contained less water at the end of the PCT analysis than the original analysis, the analysis of the three cases with a time-dependent volume showed that the original assumption was conservative.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Figure 15.06.05-52-1—Water Inventory in the Five Sub-Volume Core Concentrating Region



Question 15.06.05-53:

In the mass balance approach for prediction of boric acid accumulation due to steaming in the U.S. EPR core during post-LOCA cooling, the licensee assumed a certain initial boron concentration in the mixing volume.

As the predicted boron concentration buildup and time to precipitation depend on the assumed initial boron concentration, please describe how the values for the initial boron concentration have been calculated in the analyses performed. In the case of a large-break LOCA, explain how the core blowdown flashing and reflood steaming processes have been considered to develop a conservative estimate for the initial boron concentration.

This question is a follow-up to the reactor systems audit held on April 21-24, 2009.

Response to Question 15.06.05-53:

The initial concentration in the mixing volume for the boron precipitation calculation was set equal to the maximum injected boron concentration. The maximum injected boron concentration was determined by combining the following sources of borated water: two extra borating system (EBS) water storage tanks, four accumulators, the in-containment refueling water storage tank (IRWST), and the reactor coolant system (RCS). The concentrations and volumes of each of these sources were taken at their maximum or minimum values so as to determine the maximum possible initial boron concentration. This value was determined to be 1929 ppm, 37 percent enriched.

The actual concentration in the initial volumes (B_{init}) can be estimated based on the average enthalpy of the water circulating in the RCS just prior to the loss-of-coolant accident (LOCA). At an average temperature of 594.4°F and pressure of 2250 psia the enthalpy is 605.1 BTU/lbm. The fraction of water that evaporates is a function of the enthalpy of the water at the end of blowdown and the latent heat of evaporation:

$$X = (h_{RCS,\ pre\text{-}LOCA} - h_f) / h_{fg}$$
$$B_{init} = B_{RCS,\ pre\text{-}LOCA} / (1 - X)$$

Where

X = the quality of the expanded fluid, lbm steam / lbm total
$h_{RCS,\ pre\text{-}LOCA}$ = enthalpy of the RCS prior to the LOCA, 605.1 BTU/lbm
$B_{RCS,\ pre\text{-}LOCA}$= boron concentration of RCS prior to the LOCA
h_f = water enthalpy, BTU/lbm
h_{fg} = latent heat of evaporation, BTU/lbm

The initial concentration is calculated for three assumed post-LOCA pressures, as shown in Table 15.06.05-53-1, assuming the maximum 18 month cycle RCS operating concentration of 855 ppm. Table 15.06.05-53-1 also shows the use of 1929 ppm as the initial concentration is conservative.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Table 15.06.05-53-1—Calculated Post-LOCA Initial Boron Concentrations

Pressure	h_f	h_{fg}	X	B_{init}
psia	**BTU/lbm**	**BTU/lbm**	**lbm steam/lbm total**	**ppm, 37% enriched**
60	262	915.4	0.37	1368
45	243	929	0.39	1400
14.7	180.2	970.3	0.44	1521

Question 15.06.05-54:

In the current U.S. EPR post-LOCA boron precipitation analysis provided by the licensee, the control (mixing) volume applied includes the core fluid volume, part of upper plenum, part of lower plenum as well as the fluid volume in the guide tubes and heavy reflector.

The staff questions the applicability of this approach, as it inherently assumes that the averaged boron concentration over the entire control volume is equal to or greater than the maximum boron concentration in the core fluid volume. At this point, no quantifiable justification has been provided to substantiate the above assumption. If no quantifiable justification can be submitted, the analysis approach taken should be revised to conservatively evaluate the boron concentration and precipitation in the core where the decay heat is released. In this case, please provide the results of the revised analysis approach and discuss its impact on hot leg switch-over timing and identify all the areas to be impacted by different boron precipitation timing.

This question is a follow-up to the reactor systems audit held on April 21-24, 2009.

Response to Question 15.06.05-54:

An average concentration has regions with concentrations greater than the average as well as regions with concentrations lower than the average. The lowest boron concentration in the fuel region of the core is at the core inlet because of the emergency core cooling system (ECCS) water which provides makeup for steam generation. As this water moves up through the fuel assemblies it reaches the boiling point. There is no increase in the boron concentration in the water up to this point. As the water continues up through the core, some water is converted to steam. This increases the boron concentration in the water. It reaches the maximum concentration in the up-flow at the top of the core. This maximum boron concentration is the concentration of the water in the upper plenum and the water returning to the lower plenum through the guide tubes and the heavy reflector after recirculation is established (see the Response to RAI 15.06.05-55 and Figure 15.06.05-55-2).

The only region where the concentration could be increased above the upper plenum concentration is in the peripheral fuel assemblies where a small amount of steam is generated as water flows down toward the lower plenum. Due to the use of an average boron concentration, the boron concentration in the upper plenum is inherently under-predicted. Correspondingly, the boron concentration in the downflow through the peripheral fuel assemblies, guide tubes, and heavy reflector is under-predicted. Because the recirculating water boron concentration is under-predicted, the lower plenum boron concentration is also under-predicted. Because these regions are all under-predicted, the core is left as the only region which is over-predicted. The average concentration is the actual concentration between where the boiling starts and the exit of the core. While there is clearly simplification in using an average boron concentration, there is enough margin to the limit, as demonstrated in the analysis in support of the Response to RAI 15.06.05-55, so that it is a reasonable simplification.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 15.06.05-55:

In predicting the boric acid concentration and precipitation time due to coolant evaporation in the U.S. EPR core during post-LOCA cooling, the licensee applied a mass balance approach with an assumed control (mixing) volume. The core region fluid volume represented 61% and 56% of the total assumed mixing volume in the LBLOCA and SBLOCA analyses, respectively.

As liquid coolant is evaporated within the active core region, the dissolved boron is deposited, almost entirely, in the liquid surrounding the heated fuel surface where the evaporation process occurs. Since the predicted time to precipitation is proportional to the assumed liquid content in the defined mixing volume, please provide substantiation for the inclusion of any liquid containing regions outside of the active core in the mixing volume and explain the mechanisms capable of transporting and mixing liquid within the core region, and from the core region into any additional volumes included in the mass balance mixing volume. In addition, provide evidence that the end result of these mechanisms is perfect boric acid mixing within the liquid phase in the entire control volume as assumed in the calculations.

This question is a follow-up to the reactor systems audit held on April 21-24, 2009.

Response to Question 15.06.05-55:

The boron precipitation model is a single volume consisting of five sub-volumes (i.e., the lower plenum, the heated core regions, the guide tubes, the heavy reflector, and the upper plenum up to the hot legs). Throughout the precipitation analysis, the single volume is equal to the total liquid volume in those regions for the limiting peak cladding temperature (PCT) realistic large break loss-of-coolant accident (RLBLOCA) case at the end of the PCT transient (200 seconds). The use of a constant volume throughout the precipitation analysis is sufficiently conservative to account for differences in boron concentration in each sub-volume (see the Responses to RAI 15.06.05-52 and RAI 15.06.05-54). In addition to the single, constant volume simplification, several conservative assumptions were made. These included:

- All of the boron in the water that was evaporated stays in the region (i.e., there is no carry out of boric acid due to droplet entrainment or steam solubility).
- No increased boron solubility due to other solutes.
- No increased boiling temperature due to containment pressure and boric acid concentration.
- No addition of water containing less boric acid from sources such as the chemical and volume control system (CVCS).

The five regions were selected as the concentrating volume due to the recirculation pattern which develops post-quench in a large break loss-of-coolant accident (LBLOCA) (see Figure 15.06.05-55-1 and Figure 15.06.05-55-2). Following a LBLOCA the core steams steadily at a rate corresponding to the decay heat in the reactor core. Prior to recirculation, the core is steaming and the concentration of boron in the core increases at its fastest rate. When the heavy reflector and guide tube regions are sufficiently cooled, the flow leaving the core enters the upper plenum and either exits as steam through the hot legs or recirculates to the lower plenum. The steam exits the core via the hot legs and proceeds to the steam generators where it is further heated as it cools the secondary side. The water, on the other hand, increases the static pressure head next to the core barrel. The denser water tends to accumulate over the heavy reflector and peripheral fuel assemblies. This water then proceeds down the low power peripheral fuel assemblies, heavy reflector, and guide tubes into the lower plenum where it combines with the emergency core cooling system (ECCS) makeup and returns to the inner core regions as positive core flow.

The S-RELAP5 RLBLOCA analysis of the 18-month equilibrium cycle, Case 24 was extended to 6,000 seconds. In that case the static pressure reached the point where recirculation began around 1,000 seconds. The water returning through the heavy reflector cannot cross flow into the fueled region. Instead it flows down into the lower plenum (see Figure 15.06.05-55-3) where it mixes with other flows in the lower plenum and becomes part of the flow entering the heated fuel element region. The flow entering the guide tubes behaves in a similar manner. The recirculating flow which enters into peripheral low power fuel assemblies behaves differently (see Figure 15.06.05-55-4). As the flow moves down through these assemblies some water is evaporated in the peripheral elements. The steam carries a significant fraction of the water with it as it cross flows and combines with the flow proceeding upwards into the central regions of the core. Only about 10 percent of the water that enters the top of these peripheral fuel assemblies reaches the lower plenum (see Figure 15.06.05-55-5). The flow from the lower head to the lower plenum is in agreement with the steam flow rate and the slow increase in the core region mass (see Figure 15.6.05-55-6).

During the recirculation period, a two-phase mixture exists in the upper plenum. There is, on average, 100 ft^3 of water in the region up to the hot legs. The flow of water returning from the upper plenum to the other core regions is greater than 13 ft^3/sec. So, the average resident time in the upper plenum is less than 8 seconds. With the core steaming at a rate of 60 lbm/sec and an average core mass of 40,000 lbm, the core is concentrating at a rate of 0.15% per second. Therefore the maximum gradient in the boron concentration in the upper plenum is only 1.2% (~5 ppm). Additionally, the steam flow rate entering the upper plenum is much greater than liquid flow rate thereby thoroughly mixing the region. Thus it is appropriate for the upper plenum to be included with the other regions in the core concentrating region.

A two-region boron precipitation model was developed to provide a more detailed evaluation of the boron precipitation following a LBLOCA. The first region is the lower plenum. The second region is the remainder of the original concentrating volume. The second region is therefore composed of the heated core regions, guide tubes, heavy reflector, and upper plenum up to the hot legs. Like the single-volume precipitation model, the lower plenum and the second region have an initial concentration of 1929 ppm (see the Response to RAI 15.06.05-53). The two-region model (like the single-volume model) also uses a steam generation rate that is based on the conservative decay heat model of 1.2 times the 1973 ANS 5.1 code as referenced in 10 CFR 50, Appendix K.

The concentration of boron in the two regions as a function of time is provided in Figure 15.06.05.55-7. Because there is no recirculation from the upper plenum to the lower plenum for the first 1,000 seconds, the boron concentration is constant in the lower plenum. At 1,000 seconds, the lower plenum boron concentration increases rapidly in response to the recirculating flow. Correspondingly, the boron concentration in the second region decreases, but this region has a much larger water volume, so the decrease in concentration is smaller. Following the initial impact of the recirculating flow, the concentration in both regions continues to increase with time. At 6,000 seconds the boron concentration is less than 17,000 ppm in the lower plenum and less then 22,000 ppm in the core region.

The original single-volume model calculated a core region concentration of about 31,000 ppm at 6,000 seconds. If the flow patterns continued as they were up to 6,000 seconds, the two-region model predicts that the core region would reach the 212°F mixing precipitation limit of 38,500 ppm at approximately 4.3 hours, compared to 2.2 hours with the single, constant volume model. Therefore, the original analysis is conservative and adequate for determining the bounding time to switch to simultaneous injection to prevent boron precipitation.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Figure 15.06.05.55-1—Post-Quench LBLOCA Recirculation Depiction

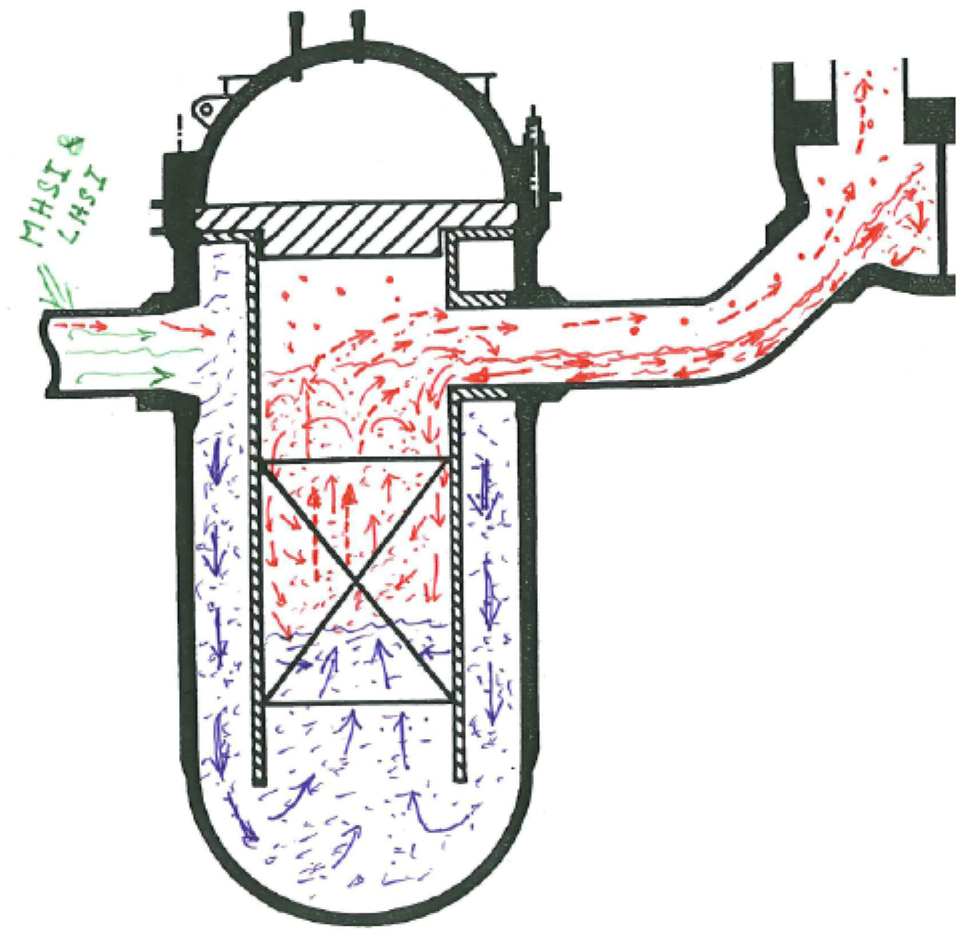


Figure 15.06.05.55-2—In-Core Pre-Recirculation and Recirculation Flow Patterns

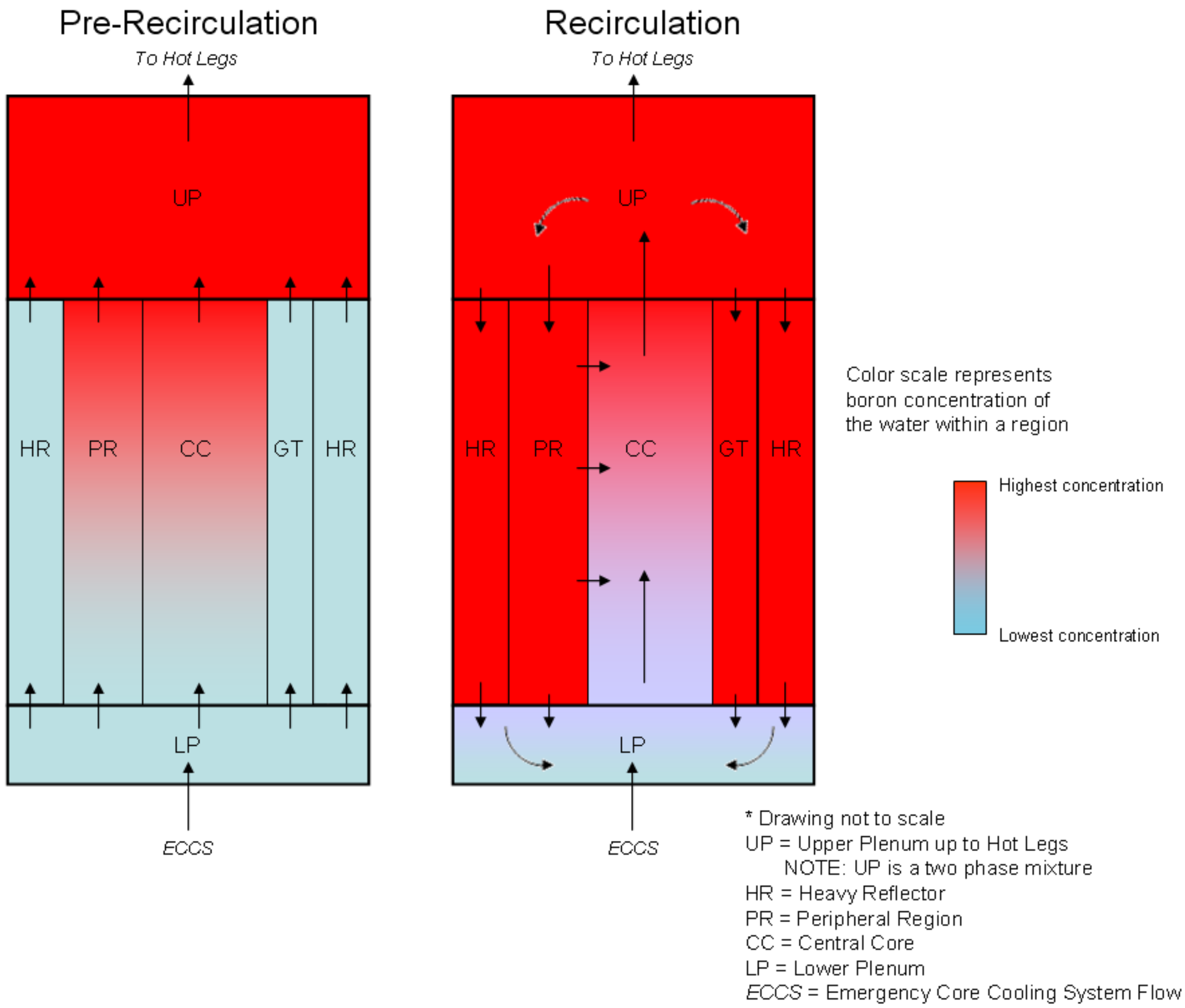


Figure 15.06.05.55-3—Integrated Flow from the Core Regions to the Upper Plenum

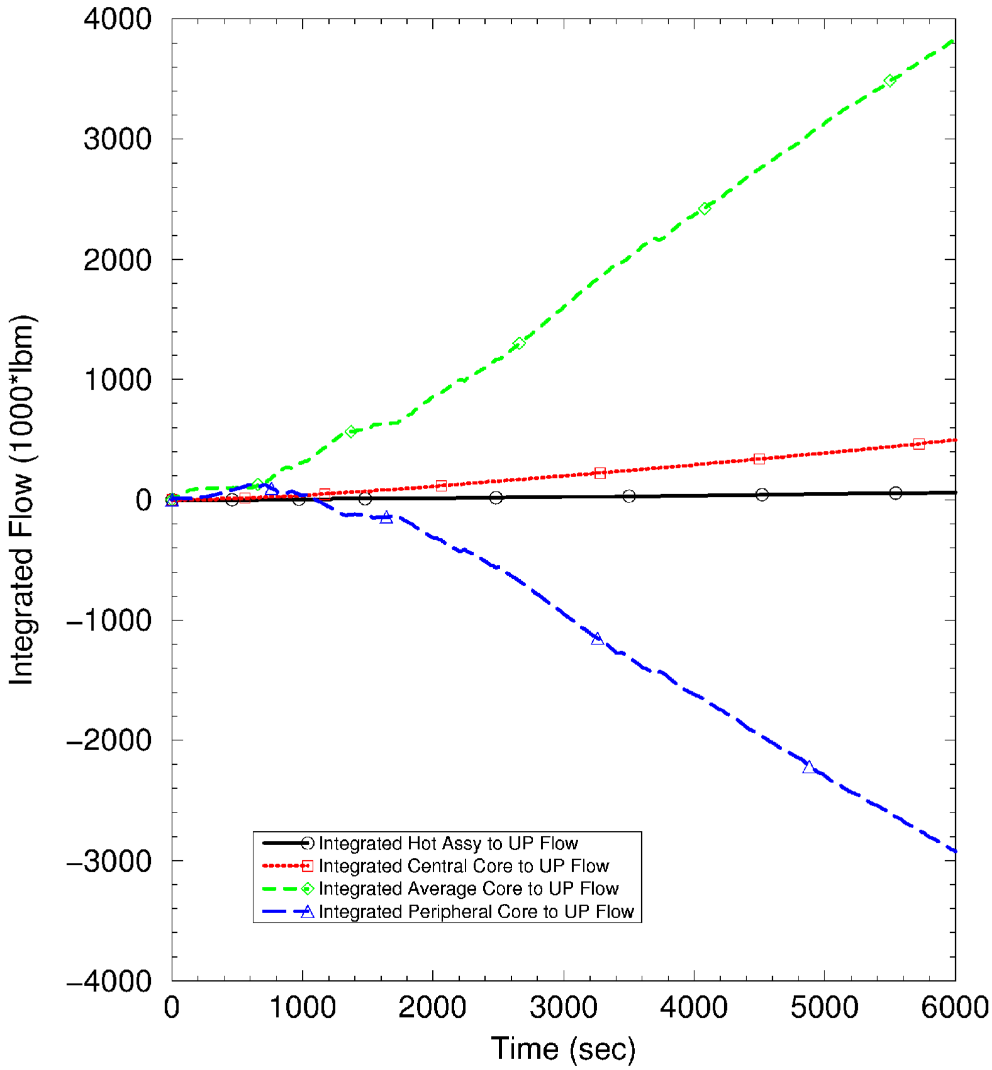


Figure 15.06.05.55-4—Integrated Flow from the Lower Plenum to the Core Regions

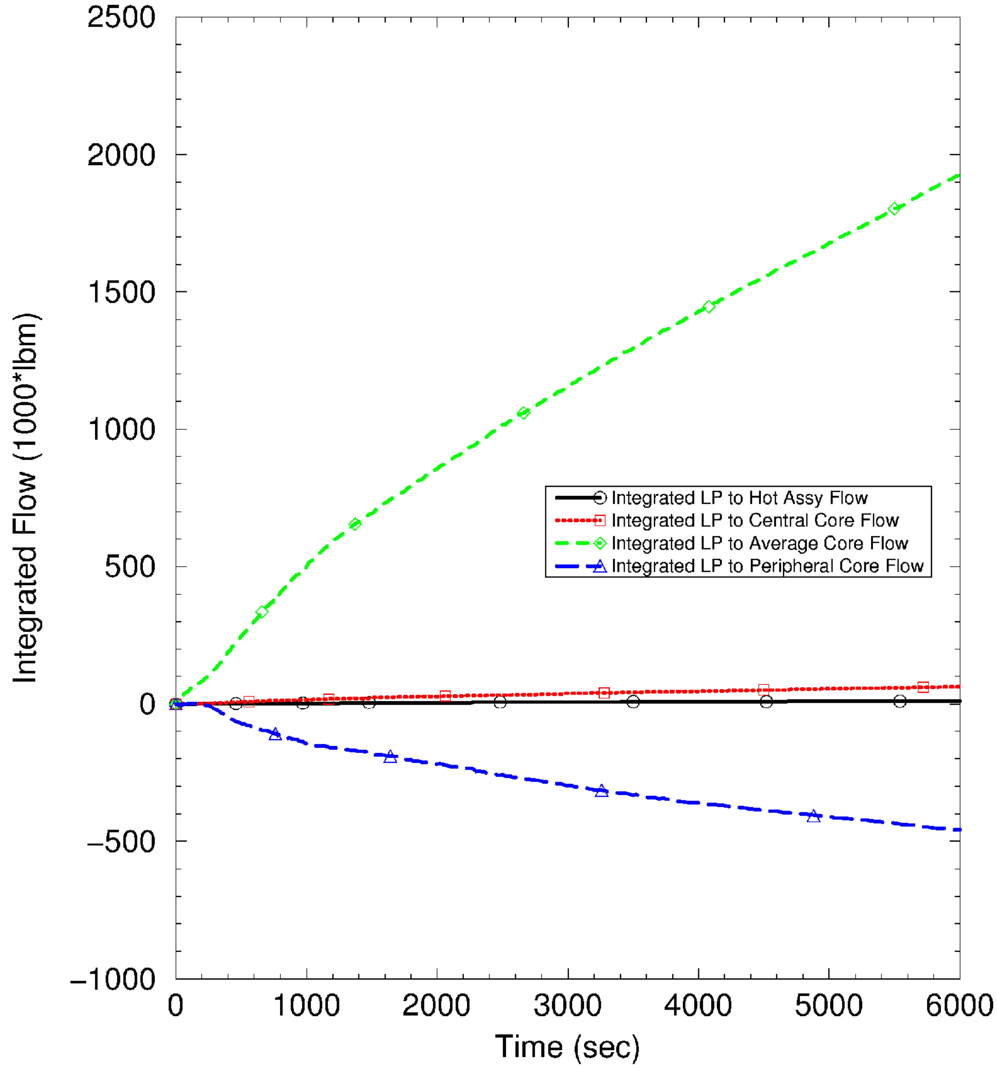


Figure 15.06.05.55-5—Integrated Flow through the Heavy Reflector and Guide Tubes

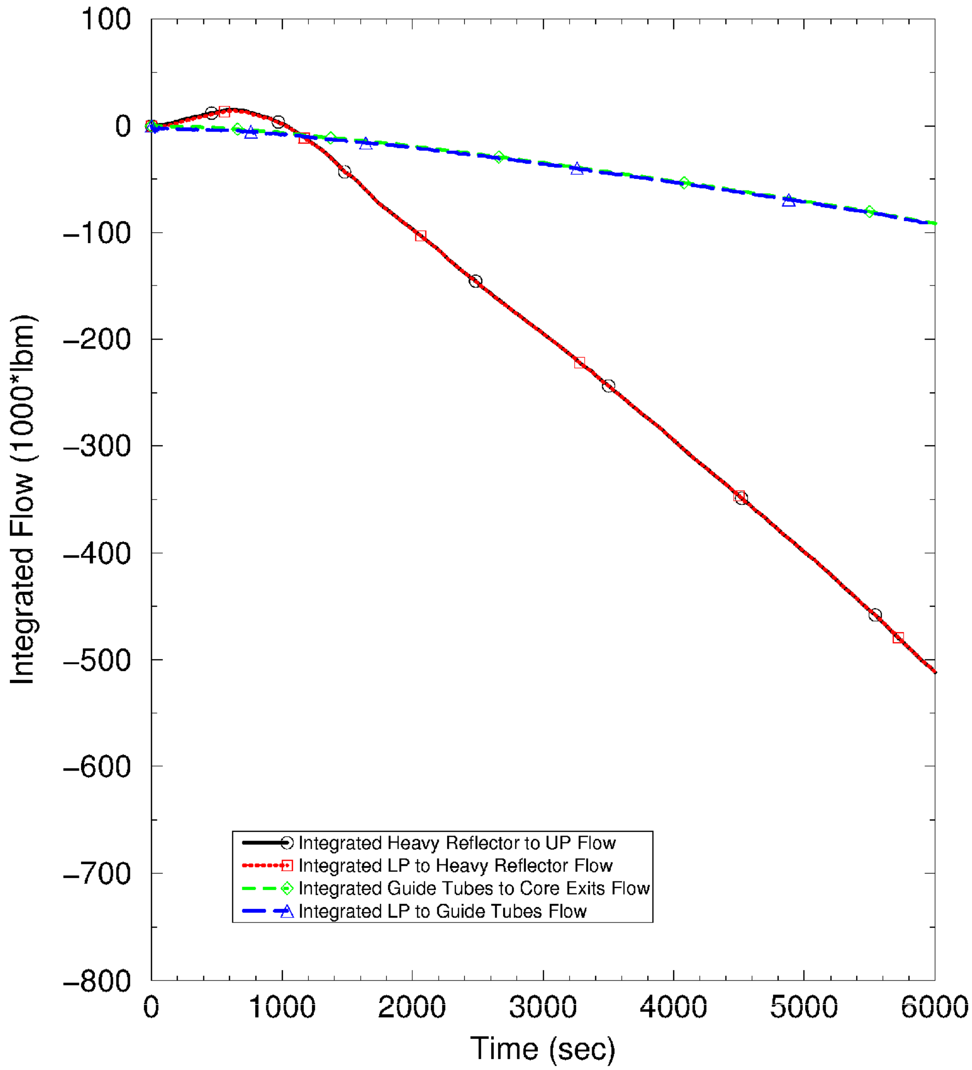


Figure 15.06.05.55-6—Integrated Flow from the Lower Head to the Lower Plenum

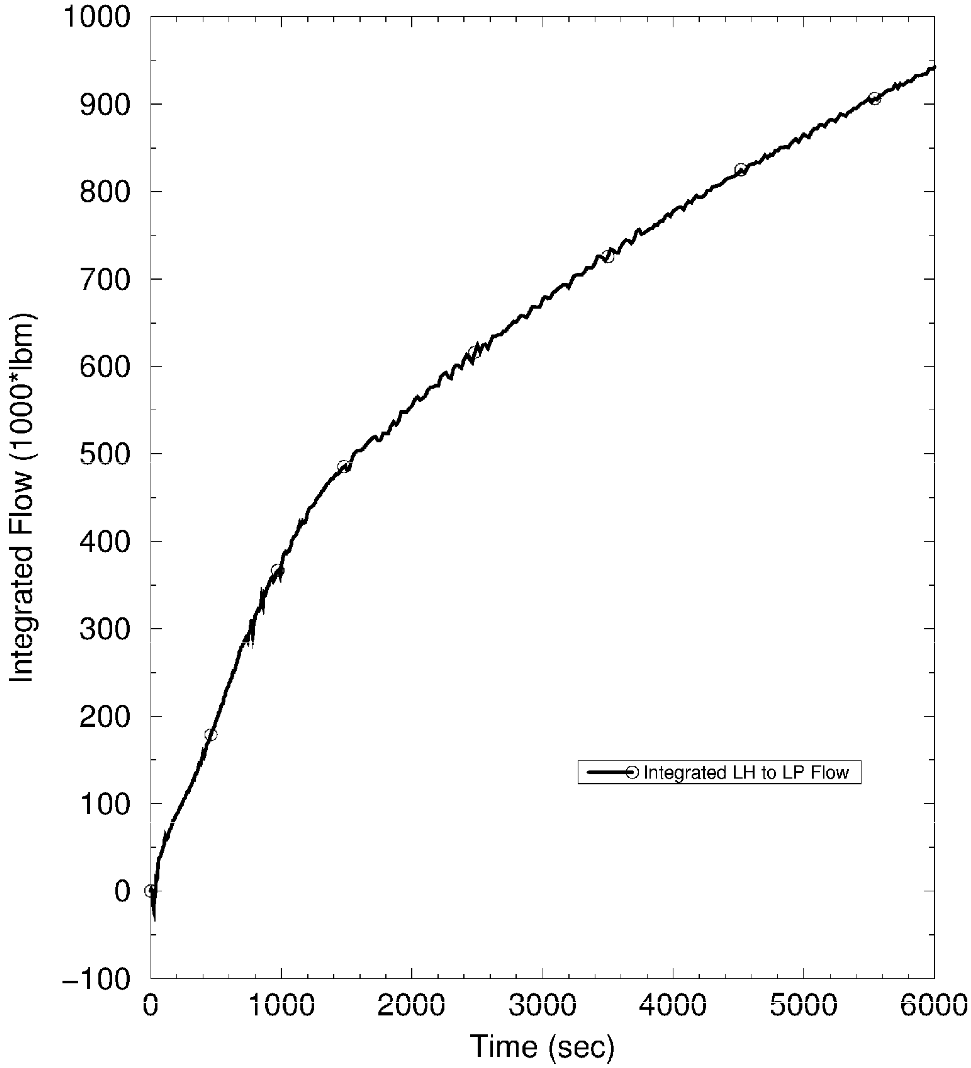


Figure 15.06.05.55-7—Boron Concentrations in the Core Regions