Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

February 11, 2011

10 CFR 50.4(b)(6)
10 CFR 50.34(b)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Watts Bar Nuclear Plant, Unit 2
NRC Docket No. 50-391

Subject: **Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Response to Requests for Additional Information**

This letter responds to requests for additional information (RAIs) regarding the Unit 2 FSAR concerning steamline break and overpressure transients.

There are no new regulatory commitments contained in this letter. If you have any questions, please contact Bill Crouch at (423) 365-2004.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 11th day of February, 2011.

Respectfully,

Marie Gillman

Marie Gillman
Acting Watts Bar Unit 2 Vice President

Enclosure:

1. Responses to RAIs Regarding Unit 2 FSAR

A053
NRR

References:

1. NRC to TVA letter dated September 16, 2010, "Watts Bar Nuclear Plant, Unit 2 - Request for Additional Information Regarding Final Safety Analysis Report Chapters 4, 5, 6 and 9 (TAC No. ME4074)" (ADAMS Accession No. ML102530464)

2. NRC to TVA letter dated September 20, 2010, "Watts Bar Nuclear Plant, Unit 2 - Request for Additional Information Regarding Final Safety Analysis Report Related to Section 15 (TAC No. ME4074)" (ADAMS Accession No. ML102290244)

3. TVA to NRC letter dated November 9, 2010, "Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Response to Requests for Additional Information" (ADAMS Accession No. ML)

cc (Enclosure):

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RAI from NRC letter dated September 16, 2010 (Reference 1)

5.2.2 - 2.a (1)

SER Section 5.0, "Reactor Coolant System and Connected Systems"

a. SER 5.2.2, "Overpressurization Protection" (FSAR 5.2.2)

(1) General Design Criterion 15 states, in part, that the reactor coolant is designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. Provide an evaluation of at-power overpressure transients, consistent with the guidelines of Section 5.2.2 of NUREG-0800.

Response

The WBN Unit 2 overpressure analyses are consistent with the requirements of Section 5.2.2 of NUREG-0800, "*Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition*." Section 5.2.2 requires that the second safety grade reactor trip signal be credited for safety valve sizing calculations. This is consistent with the safety valve sizing procedure discussed in Section 2 of WCAP-7769, "*Topical Report Overpressure Protection for Westinghouse Pressurized Water Reactors*," Revision 1, June, 1972. WCAP-7769 states, "For the sizing, main feedwater flow is maintained and no credit for reactor trip is taken." This analysis is typically performed prior to construction of the plant to provide a basis for the capacity requirements for the safety valves, and the requirement of Section 5.2.2 provides a conservative basis for the number and design of the valves.

However, WCAP-7769 goes on to say, "After determining the required safety valve relief capacities, as described above, the loss of load transient is again analyzed for the case where main feedwater flow is lost when steam flow to the turbine is lost ...For this case, the bases for analysis are the same as described above except that credit is taken for Doppler feedback and appropriate reactor trip, other than direct reactor trip on turbine trip." This describes the analysis performed in Chapter 15 of the Unit 2 FSAR, which verifies that the overpressure limits are satisfied with the current/latest design.

The analyses documented in the WBN Unit 2 FSAR are not safety valve sizing calculations. No changes are being made to the safety valves as a result of the Unit 2 completion program. The Loss of External Electrical Load / Turbine Trip analysis demonstrates that the safety valves have adequate capacity to maintain peak primary pressure below 110% of design, which satisfies the requirements of General Design Criteria (GDC) 15. GDC-15 applies to "any condition of normal operation, including anticipated operational occurrences" which does not include a common mode failure of the first safety grade reactor trip signal.

The Loss of External Load / Turbine Trip RCS overpressure analysis is performed to demonstrate that, in the event of a sudden loss of the secondary heat sink, the associated increase in reactor coolant system temperature does not result in overpressurization of the RCS system.

ENCLOSURE 1

Response to RAIs Regarding Unit 2 FSAR

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The NRC requested the results of an additional Loss of Load analysis with no credit taken for the first safety grade trip reached. In the licensing basis Loss of Load analysis for WBN Unit 2, the first safety grade trip reached is the High Pressurizer Pressure trip. With credit for this trip function, the peak primary system pressure reached is 2691.8 psia. A new analysis with no credit taken for reactor trip via High Pressurizer Pressure has been run. In this analysis, reactor trip is provided by the Over Temperature Delta T protective function. Reactor trip occurs 4.5 seconds later than in the licensing basis case, and the new peak primary system pressure is 2714.7 psia. Thus, even with no credit for the first trip, the peak primary pressure does not exceed 110% of the design pressure (i.e., 1.1 X 2500 = 2750 psia).

ENCLOSURE 1

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RAI from NRC letter dated September 20, 2010 (Reference 2)

15.0.0 - 3.b.

FSAR 15.3.2, "Minor Secondary System Pipe Breaks"

a. *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.*

Response

A response to this RAI was previously provided in Reference 3; however, the staff reviewer verbally requested that a new response be provided. The new response follows.

The current licensing basis steamline break presented in the WBN Unit 2 FSAR consists of a 1.4 ft^2 double ended steamline break. The detailed sequence of events for the FSAR case is as follows:

Event	Time (seconds)
Break occurs	0.0
Steamline isolation occurs	8.67
Feedline isolation occurs	8.67
Safety injection starts	27.67
Reactivity feedback causes returns to critical	43.8
Accumulators actuate	53.6
Peak heat flux reached (1.6%)	57.4
Minimum DNBR reached	57.4
Subcriticality is reached	58.4

The NRC's concern is that the transient is turned around due to the injection of boron from the accumulators and that a smaller break may delay accumulator actuation such that a smaller break could be more limiting. The following table shows how the timing of accumulator actuation as well as the timing and magnitude of the peak heat flux and the calculated minimum DNBR change with break size.

Break size	Time of accumulator actuation (seconds)	Peak heat flux (1)	Time of peak heat flux (seconds)	Improvement in the Minimum DNBR (3)
1.4 ft^2	53.6	1.6%	57.4	(2)
1.2 ft^2	59.8	1.3%	64.4	+21%
1.0 ft^2	67.0	1.1%	71.8	+41%
0.8 ft^2	76.8	0.9%	81.8	+71%
0.6 ft^2 and smaller	No return to power, so no DNBR calculation is done.			

(1) Peak heat flux after shutdown margin is lost, and core reactivity is greater than zero.

(2) Due to the benign nature of the WBN hot zero power steamline break (HZP SLB) transient as indicated by the peak heat flux reached, the minimum DNBR for the limiting case (1.4 ft^2) is very large (>10), well above the W-3 low pressure correlation limit of 1.45.

(3) This is the improvement in the calculated minimum DNBR relative to the calculated minimum DNBR for the 1.4 ft^2 break case. For example, the calculated minimum DNBR for the 1.2 ft^2 break is 21% larger than the calculated minimum DNBR for the 1.4 ft^2 break.

Other results showing the impact of break size on the severity of the event are as follows:

Break size	Time of accumulator actuation (seconds)	Time of Low Steamline Pressure * (seconds)	Maximum core reactivity (pcm)	Time of peak core reactivity (seconds)
1.4 ft^2	53.6	0.668	120.80	54.40
1.2 ft^2	59.8	0.758	96.11	60.60
1.0 ft^2	67.0	0.878	63.76	68.00
0.8 ft^2	76.8	1.043	21.72	77.80
0.6 ft^2	90.8	1.286	-30.80	91.80
0.4 ft^2	112.2	1.674	-104.6	113.2
0.2 ft^2	152.2	2.403	-213.3	153.2
0.1 ft^2	230.7	3.609	-337.1	231.2
0.05 ft^2	317.2	4.192	-356.1	303.2

* Safety injection (SI), feedline isolation (FLI) and steamline isolation (SLI) all actuate via this signal – FLI and SLI actuate 8 seconds later, and SI actuates 27 seconds later. Note that SLI isolated three steam generators. It is conservatively assumed that the break is located between the steam generator and the isolation valve.

Transient plots that further demonstrate transient severity variability with break size are included on the following pages. Figure 1 shows Nuclear Power versus time for each of the cases run. The largest break shows the highest nuclear power with each successive case falling below the large break case. Figure 2 is core heat flux versus time. The trends seen are identical to the Nuclear Power curve. Figure 3 is RCS pressure versus time. As expected, the RCS pressure drops the most for the largest break. Figure 4 shows core reactivity versus time. The largest break case has the highest peak, and then drops the fastest due to the earlier accumulator and safety injection actuations, while the other cases follow as expected. Note that only the 4 largest breaks return to power (reactivity > 0.0). Figure 5 shows core boron concentration versus time. The largest break has the earliest and fastest increase in boron concentration due to the earlier accumulator and SI actuations and the lower RCS pressures (SI capacity is a function of pressure). The other cases follow as expected.

In conclusion, the provided tables and figures clearly show that the severity of the transient decreases as the break size decreases. The limiting case is the largest break analyzed which corresponds to the area of the flow restrictor located in the steam generator outlet nozzle (i.e., 1.4 ft^2).

ENCLOSURE 1

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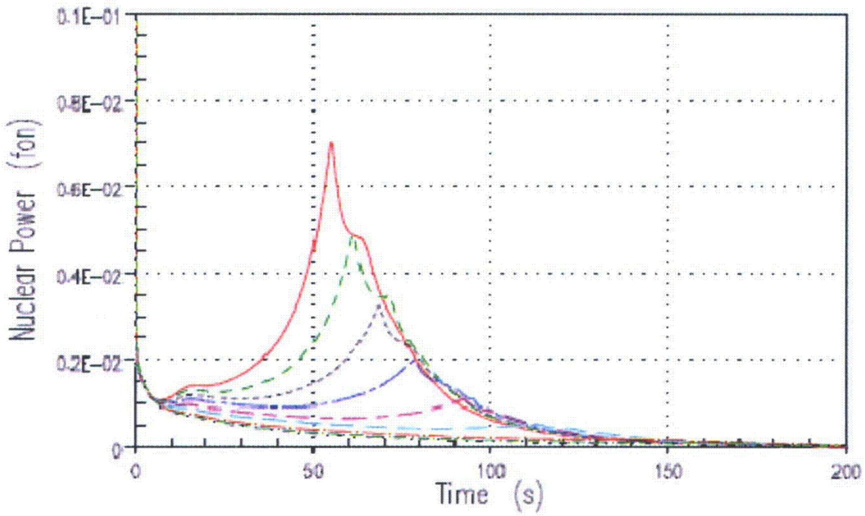

Figure 1 - Nuclear Power vs. Time

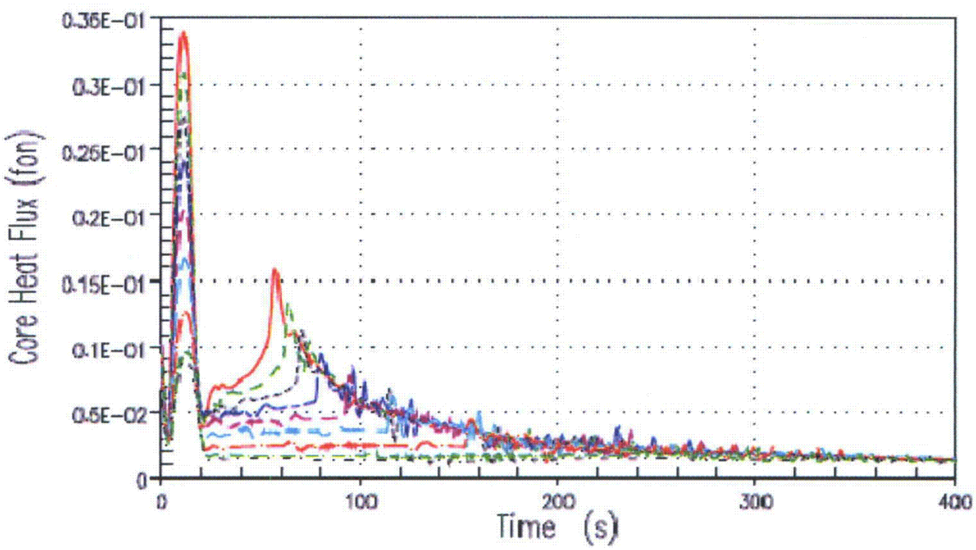

Figure 2 - Core Heat Flux vs. Time

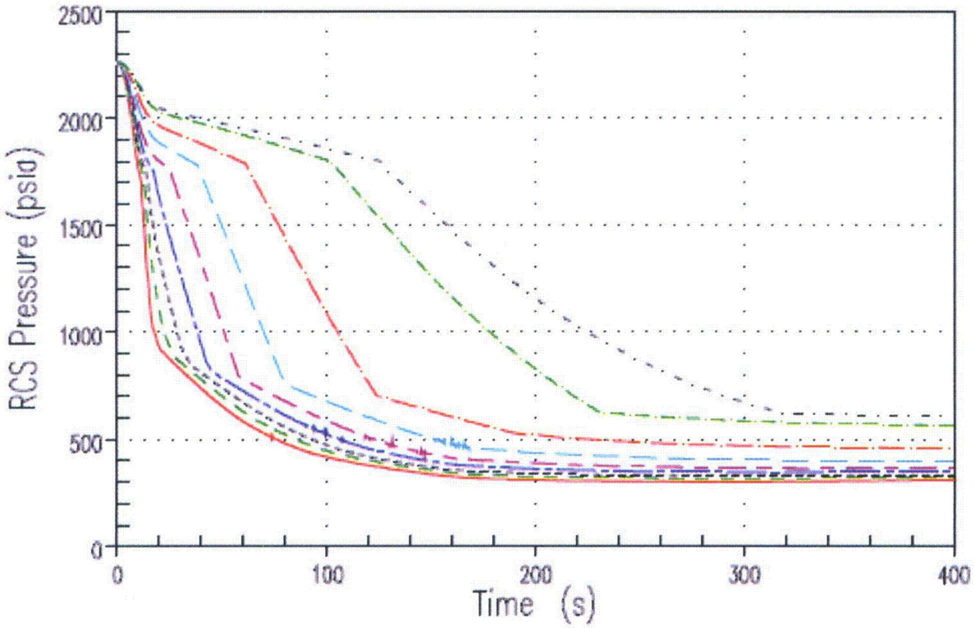

Figure 3 - RCS Pressure vs. Time

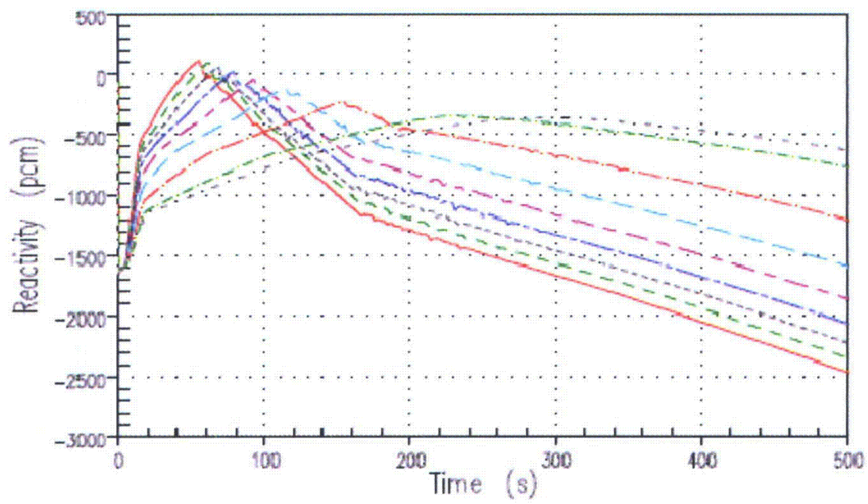

Figure 4 - Core Reactivity vs. Time

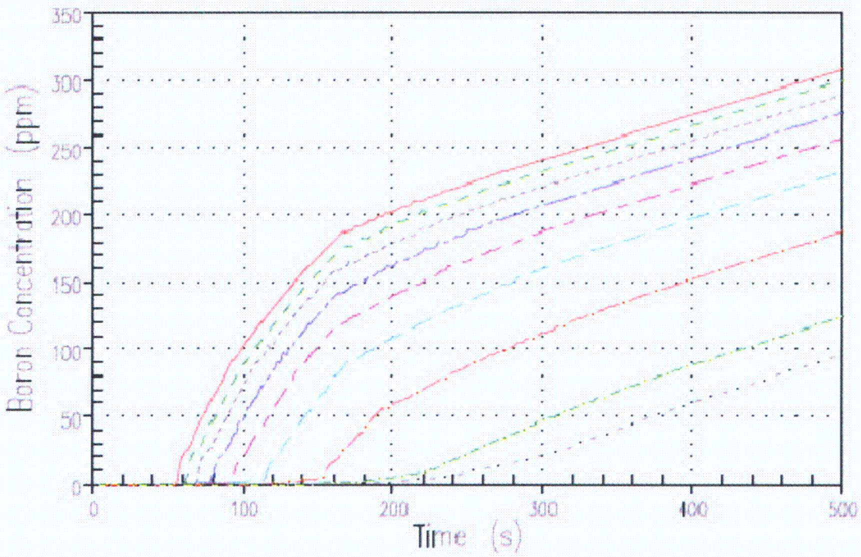

Figure 5 - Core Boron Concentration vs. Time