WBN2Public Resource

From: Boyd, Desiree L [dlboyd@tva.gov]
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To: Epperson, Dan; Wilson, George; Poole, Justin; Milano, Patrick
Cc: Arent, Gordon; Hamill, Carol L; Boyd, Desiree L
Subject: TVA letter to NRC_06-27-12_FSAR Chapter 15 Commitment Closure
Attachments: 06-27-12_FSAR Chapter 15 Commitment Closure_Final.pdf

Please see attached TVA letter that was sent to the NRC today.

Thank You,

~*~*~*~*~*~*~*~*~*~*~*~

Désireé L. Boyd

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10 CFR 50.34(b)

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

Subject: Watts Bar Nuclear Plant (WBN) Unit 2 – Submittal of Non-Proprietary Information Related to Final Safety Analysis Report Chapter 15 Transient Analysis

Reference: TVA letter to NRC dated December 10, 2010, "Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Response to Request for Additional Information"

The referenced letter provided TVA's responses to requests for additional information (RAI) on a number of sections of the FSAR. Enclosure 4 of the referenced letter contained Westinghouse proprietary information related to RAIs on the Chapter 15 transient analyses. Commitment 14 from Enclosure 6 of the reference documented TVA's agreement to provide a non-proprietary version of Enclosure 4 and the associated withholding affidavit.

Enclosure 1 of this letter provides the non-proprietary version of Enclosure 4 and Enclosure 2 provides the withholding affidavit. This submittal completes the actions from Commitment 14. The information provided in this letter does not supersede any information submitted to the NRC after December 10, 2010.

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

Respectfully,

R.A. Hruby, Jr.

Raymond A. Hruby, Jr.
General Manager, Technical Services
Watts Bar Unit 2

Enclosures:

1. Westinghouse Proprietary Review of Enclosure 4 of TVA RAI Response (T02 101210 001), “Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Response to Requests for Additional Information,” Dec. 10, 2010

2. Affidavit CAW-12-336, “Application for Withholding Proprietary Information from Public Disclosure”

cc (Enclosures):

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ENCLOSURE 1

Westinghouse Proprietary Review of Enclosure 4 of TVA RAI Response (T02 101210 001), "Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Response to Requests for Additional Information," Dec. 10, 2010

Westinghouse Proprietary Review of Enclosure 4 of TVA RAI Response (T02 101210 001, "Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Response to Requests for Additional Information," Dec. 10, 2010)

Westinghouse Electric Company LLC
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15.0.0 – 1. *FSAR 15.0.0, "Accident Analyses"*

b. *For accident analyses and evaluations in Chapter 15, identify (a) the event classification - Condition II, III, or IV, (b) the specific analysis acceptance criteria to be satisfied for that event, and (c) how each of the criteria are satisfied. If the event is considered to be enveloped by another event, explain the bases and comparisons that underlie the conclusion.*

Response:

The ANS classification of plant conditions divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

Condition I: Normal Operation and Operational Transients

Condition II: Faults of Moderate Frequency

Condition III: Infrequent Faults

Condition IV: Limiting Faults

The RAI requested specified information be provided for Condition II, III, and IV events.

Condition II: Faults of Moderate Frequency are described in Section 15.2 of the Unit 2 FSAR. This section starts as follows:

"15.2 CONDITION II - FAULTS OF MODERATE FREQUENCY

These faults, at worst, result in the reactor trip with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., Condition III or IV category. In addition, Condition II events are not expected to result in fuel rod failures or reactor coolant system (RCS) overpressurization. For the purposes of this report, the following faults have been grouped into this category:

(1) Uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition.

(2) Uncontrolled rod cluster control assembly bank withdrawal at power.

(3) Rod cluster control assembly misalignment.

(4) Uncontrolled boron dilution.

(5) Partial loss of forced reactor coolant flow.

(6) Startup of an inactive reactor coolant loop.

(7) Loss of external electrical load and/or turbine trip.

(8) Loss of normal feedwater.

(9) Loss of offsite power to the station auxiliaries (station blackout).

(10) Excessive heat removal due to feedwater system malfunctions.

(11) Excessive load increase incident.

(12) Accidental depressurization of the reactor coolant system.

(13) Accidental depressurization of the main steam system.

(14) Inadvertent operation of emergency core cooling system during power operation.

An evaluation of the reliability of the reactor protection system actuation following initiation of Condition II events is presented ... for the relay protection logic. Standard reliability engineering techniques were used to assess likelihood of the trip failure due to random component failures. Common mode failures were also qualitatively investigated. It was concluded from the evaluation that the likelihood of no trip following initiation of Condition II events is extremely small (2×10^{-7} derived for random component failures). The solid state protection system design has been evaluated by the same methods as used for the relay system and the same order of magnitude of reliability is provided.

The worst common mode failure which is postulated to occur is the failure to scram the reactor after an anticipated transient has occurred. A series of generic studies, ... on anticipated transients without scram (ATWS) showed acceptable consequences would result provided that the turbine trips and auxiliary feedwater flow is initiated in a timely manner. The effects of ATWS events are not considered as part of the design basis for transients analyzed in Chapter 15. The final NRC ATWS rule [12] requires that Westinghouse-designed plants install ATWS mitigation system circuitry (AMSAC) to initiate a turbine trip and actuate auxiliary feedwater flow independent of the reactor protection system. The Watts Bar AMSAC design is described in Section 7.7.1.12."

The information requested for each event follows:

(1) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition

a. Condition II

b. Acceptance Criteria

Fuel cladding integrity should be maintained by ensuring that the minimum DNBR remains above the 95/95 limit, based on an acceptable correlation. The safety analysis DNBR limits based on the W-3 correlation for regions below the mixing vane grids (first span) and the WRB-1 correlation for the regions between mixing vane grids (remaining spans) are as follow.

Below mixing vane grid, first span (W-3): [ ]^{a,c} *

Between mixing vane grids, remaining spans (WRB-1): [ ]^{a,c} *

The fuel temperature limit should not be exceeded. The peak linear heat generation rate (expressed in kw/ft) should not exceed a value which would cause fuel centerline melt. The limit for fuel centerline is 4800°F. The design basis peak linear heat generation rate is not to exceed 18.0 kw/ft.

c. Analysis Results

1. A DNB statepoint evaluation showed that the DNBR remained above the safety limit DNBR value for the uncontrolled RCCA bank withdrawal from subcritical event. No fuel damage is predicted to occur since the DNBR remained above the limit value. The most limiting statepoint for this event is as follows:

- Core Inlet Coolant Temperature: 557.0°F
- Nominal Core Heat Flux: 189,800.0 Btu/ft^2-hr
- Core Pressure: 2200.0 psia
- Core Mass Flow Rate: 1,130,646.0 lbm/hr-ft^2 (46% of Thermal Design Flow)
- Time: 12.40 (seconds) ; Heat Flux: 0.3712 (Fraction of Nominal)

The calculated minimum DNBR for the most limiting statepoint is presented below.

Fuel Assembly

Region	MDNBR Limit*	MDNBR Calculated
First span:	[ ]^{a,c}	2.064
Remaining spans:	[ ]^{a,c}	1.950

2. The maximum fuel centerline temperature was found to be 2234°F, compared to the limit value of 4800°F.

* Minimum DNBR limits are based on the V5H fuel type and corresponding W-3 DNB correlation (first span) and WRB-1 DNB correlation (remaining spans). This event was evaluated for the RFA-2 with the Intermediate Fuel Mixing grids Fuel Upgrade Program. It was demonstrated that the minimum DNBR limit continues to be met, corresponding to the RFA-2 with IFMs fuel-specific W-3 DNB correlation limit (first span) and the WRB-2M DNB correlation (remaining spans).

(2) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (RWAP)

a. Condition II

b. Acceptance Criteria

Pressure in the Reactor Coolant System and Main Steam System should be maintained below 110% of the design values. (Note: The pressurization of the Reactor Coolant System and Main Steam System due to the RWAP event is bounded by the Loss of Load/Turbine Trip event.)

Fuel cladding integrity should be maintained by ensuring that the minimum DNBR remains above the 95/95 limit, based on an acceptable correlation. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. Westinghouse demonstrates that this criterion is met for the RWAP event by showing that the pressurizer does not become water solid.

c. Analysis Results

The most limiting statepoint occurred for the 60% power RWAP case with minimum feedback and a reactivity insertion rate of 8 pcm/sec. A DNB calculation for the limiting statepoint was not required.

The calculated peak pressurizer water volume is 1698 ft^3, demonstrating that the pressurizer does not fill as a result of an RWAP event.

As mentioned previously, the results of the RWAP analysis show that the 60% power case with minimum feedback and reactivity insertion rate of 8 pcm/sec was the most limiting RWAP even for the DNB transient. The DNB evaluation showed that the minimum DNBR remained above the safety limit value. No fuel damage is predicted to occur since the DNBR remained above the limit value.

The calculated minimum DNBR for the limiting RWAP case is as follows:

Case	MDNBR Limit	MDNBR Calculated
RWAP - 60% power, minimum feedback, 8 pcm/sec reactivity insertion rate	[ ]^{a,c}	1.417

(3) Rod Cluster Control Assembly Misalignment

a. Condition II

b. Acceptance Criteria

Pressure in the Reactor Coolant System and Main Steam System should be maintained below 110% of the design values.

Fuel clad integrity should be maintained by ensuring that the minimum DNBR remains above the 95/95 limit, based on an acceptable correlation.

An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

c. Analysis Results

No fuel damage occurred since the DNBR remained above the limit value.

(4) Uncontrolled Boron Dilution

a. Condition II

b. Acceptance Criteria

Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.

Fuel cladding should be maintained by ensuring that the minimum DNBR remains above the 95/95 limit, based on an acceptable correlation.

An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

If operator action is required to terminate the transient, the following minimum intervals must be available between the initiation of the uncontrolled boron dilution event and the time of complete loss of shutdown margin:

a. Refueling (Mode 6): 30 minutes

b. Startup and Power (Modes 2 and 1): 15 minutes

c. Analysis Results

For the Boron Dilution at Power (Mode 1), it was calculated that 33.0 minutes (automatic rod control) and 34.3 minutes (manual rod control) are available from beginning of the event until loss shutdown margin, which meets the acceptance criterion of at least 15 minutes.

For the Boron Dilution During Startup (Mode 2), it was calculated that 26.4 minutes are available from beginning of the event until loss shutdown margin, which meets the acceptance criterion of at least 15 minutes.

(5) Partial Loss of Forced Reactor Coolant Flow (PLOF)

a. Condition II

b. Acceptance Criteria

Pressure in the Reactor Coolant System and Main Steam System should be maintained below 110% of the design values. (Note: The pressurization of the Main Steam System due to a loss of flow event is bounded by the Loss of Load/Turbine Trip event.)

Fuel clad integrity should be maintained by ensuring that the minimum DNBR remains above the 95/95 limit, based on an acceptable correlation.

An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

c. Analysis Results

The most limiting statepoint for the Partial Loss of Flow analysis is as follows:

- Initial Core Average Heat Flux: 0.192455 x 10^6 Btu/hr-ft^2
- Initial Volumetric Flow: 379,100 gpm
- Core Inlet Temperature: 557.3°F
- Pressurizer Pressure: 2250 psia

Time (Seconds)	Heat Flux (Fraction of Initial)	Mass Flow Rate (Fraction of Initial)
3.70	0.994	0.935

The calculated peak RCS pressure is 2412 psia, demonstrating that the Reactor Coolant System remains below 110% of design.

The DNB evaluation showed that the minimum DNBR remained above the safety limit DNBR value. No fuel damage is predicted to occur since the DNBR remained above the limit value. The calculated minimum DNBR follows:

Case	MDNBR Limit*	MDNBR Calculated
PLOF	[ ]a,c	1.763

* Minimum DNBR limits are based on the V5H fuel type and corresponding WRB-1 DNB correlation. This event was evaluated for the RFA-2 with the Intermediate Fuel Mixing grids Fuel Upgrade Program. It was demonstrated that the minimum DNBR limit continues to be met, corresponding to the WRB-2M DNB correlation.

(6) Startup of an Inactive Reactor Coolant Loop

a. Condition II

b. Acceptance Criteria

Pressure in the Reactor Coolant System and Main Steam System should be maintained below 110% of the design values.

Fuel cladding should be maintained by ensuring that the minimum DNBR remains above the 95/95 limit, based on an acceptable correlation.

An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

c. Analysis Results

The startup of an inactive coolant loop event was considered in the original design basis for WBN when potential operation with a loop out of service (i.e., N-1 loop operation) was considered. However, subsequent to initial plant startup, N-1 loop operation was not pursued. Plant operation at power with a loop out of service is precluded by the Technical Specifications which require that all four reactor coolant pumps be operating. Therefore, since at power initial conditions associated with N-1 loop operation are prohibited, no analysis is required to show that the analysis criteria are satisfied for this event.

(7) Loss of External Electrical Load and/or Turbine Trip

a. Condition II

b. Acceptance Criteria

Pressure in the RCS and Main Steam System should be maintained below 110% of the design values.

Fuel cladding should be maintained by ensuring that the minimum DNBR remains above the 95/95 limit, based on an acceptable correlation.

An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

c. Analysis Results

The calculated peak pressurizer and steam generator pressures are 2652 psia and 1281 psia respectively, demonstrating that the RCS and Main Steam System remain below 110% of design.

No fuel damage occurred since the DNBR remained above the limit value.* The calculated minimum DNBRs for all both cases are as follows:

Case	MDNBR
Minimum reactivity feedback with pressure control	1.538 at t = 12.6 sec
Minimum reactivity feedback without pressure control	1.905 at t = 0.0 sec

* Minimum DNBR limits are based on the V5H fuel type and corresponding WRB-1 DNB correlation. This event was evaluated for the RFA-2 with the Intermediate Fuel Mixing grids Fuel Upgrade Program. It was demonstrated that the minimum DNBR limit continues to be met, corresponding to the WRB-2M DNB correlation.

(8) Loss of Normal Feedwater (LONF)

a. Condition II

b. Acceptance Criteria

Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.

Fuel cladding should be maintained by ensuring that the minimum DNBR remains above the 95/95 limit, based on an acceptable correlation.

An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

With respect to peak pressures, the LONF and LOOP events are bounded by the loss of load/turbine trip analysis.

With respect to DNB, the LONF event is bounded by the loss of load/turbine trip analysis. The only difference between these two events is turbine trip which is not assumed in a LONF event until after reactor trip. This allows for continued heat removal (steam flow), which is a benefit, until rod motion. The LOOP event is bounded by the complete loss of flow analysis. The flow coastdown in the LOOP event does not occur until after reactor trip, which is less limiting. Therefore, the event is not analyzed for DNB concerns, but rather, for the long-term heat removal capability.

For ease in interpreting the transient results following a loss of normal feedwater, Westinghouse has adopted the following restrictive acceptance criterion:

Westinghouse Acceptance Criteria:

The criterion that the pressurizer shall not become water solid is employed to ensure that a more serious plant condition is not generated as a result of a LONF or LOOP event.

c. Analysis Results

The RCS and main steam system remain below 110% of design pressure.

LONF:

The pressurizer does not become water solid. The peak pressurizer water volume (1,352* ft^3) remains below the available pressurizer volume (1,841 ft^3) which includes the pressurizer surge volume.

LOOP:

The pressurizer does not become water solid. The peak pressurizer water volume (1,472* ft^3) remains below the available pressurizer volume (1,841 ft^3) which includes the pressurizer surge volume.

* Maximum pressurizer water volumes given are the "second peak," which occurs later in the LONF/LOOP transient. The second peak is of primary importance since the LONF/LOOP events are analyzed to demonstrate long-term heat removal capability of the auxiliary feedwater system. Note, however, that the "first peak" volumes also remain below the available pressurizer volume (1,841 ft^3) for the LONF and LOOP transients.

(9) Loss of offsite power to the station auxiliaries (station blackout).

a. **Condition II**

b. **Acceptance Criteria**

See **Acceptance Criteria** for **item (7)** on page E4 - 7.

c. **Analysis Results**

See **Analysis Results** for **item (7)** on pages E4 - 7 and E4 - 8.

(10) Excessive Heat Removal due to Feedwater System Malfunction

a. **Condition II**

b. **Acceptance Criteria**

Pressure in the Reactor Coolant System and Main Steam System should be maintained below 110 % of the design values. (Note: The pressurization of the Reactor Coolant System and Main Steam System due to Feedwater Malfunction event is bounded by the Loss of Load/Turbine Trip event.)

Fuel cladding should be maintained by ensuring that the minimum DNBR remains above the 95/95 limit, based on an acceptable correlation.

An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

c. **Analysis Results**

For the four Hot Full Power cases, the following minimum DNBR values were determined:

Case	Minimum DNBR
Hot Full Power, Single-Loop FWM, automatic rod control	1.653 @ 52.0 seconds
Hot Full Power, Single-Loop FWM, manual rod control	1.579 @ 26.5 seconds
Hot Full Power, Multi-Loop FWM, automatic rod control	1.511 @ 16.5 seconds
Hot Full Power, Multi-Loop FWM, manual rod control	1.516 @ 25.5 seconds

Since the calculated minimum DNBRs are above the DNBR safety analysis limit [ ]^{a,c} *, no fuel or cladding damage is predicted.

The cases of an accidental full opening of one or more feedwater control valves with the reactor at hot zero power are bounded by the hot full power cases as mentioned above. Therefore, the results of the analyses are not presented.

* The minimum DNBR limit is based on the V5H fuel type and corresponding WRB-1 DNB correlation. This event was evaluated for the RFA-2 with the Intermediate Fuel Mixing grids Fuel Upgrade Program. It was demonstrated that the minimum DNBR limit continues to be met, corresponding to the WRB-2M DNB correlation.

(11) Excessive Load Increase Incident

a. Condition II

b. Acceptance Criteria

Pressure in the RCS and Main Steam System should be maintained below 110% of the design values. Fuel cladding should be maintained by ensuring that the minimum DNBR remains above the 95/95 limit, based on an acceptable correlation.

An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

c. Analysis Results

No fuel damage occurred since the DNBR remained above the safety analysis limit value.

Minimum reactivity feedback with manual rod control: [ ][a,c]

Maximum reactivity feedback with manual rod control: [ ][a,c]

Minimum reactivity feedback with automatic rod control: [ ][a,c]

Maximum reactivity feedback with automatic rod control: [ ][a,c]

The analysis method employed compares conservative RCS conditions resulting from the transient (i.e., core power, T_{avg}, pressurizer pressure, and flow) with the plant core thermal limits. Hence, the analysis demonstrates that the DNBR design basis is met without explicitly calculating the minimum DNBR value. The conservative RCS conditions are based on historical analysis performed for the Excessive Load Increase event.

(12) Accidental Depressurization of the Reactor Coolant System

a. Condition II

b. Acceptance Criteria

Pressure in the RCS and Main Steam System should be maintained below 110% of the design values.

Fuel cladding should be maintained by ensuring that the minimum DNBR remains above the 95/95 limit, based on an acceptable correlation.

An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

c. Analysis Results

No fuel damage occurred since the DNBR remained above the safety analysis limit value.* The calculated minimum DNBR for the case analyzed is 1.561.

* Minimum DNBR limits are based on the V5H fuel type and corresponding WRB-1 DNB correlation. This event was evaluated for the RFA-2 with the Intermediate Fuel Mixing grids Fuel Upgrade Program. It was demonstrated that the minimum DNBR limit continues to be met, corresponding to the WRB-2M DNB correlation.

(13) Accidental Depressurization of the Main Steam System

a. Condition II

b. Acceptance Criteria

For the Credible Break (Condition II):

Pressure in the RCS and Main Steam System should be maintained below 110% of the design values.

Fuel cladding, should be maintained by ensuring that the minimum DNBR remains above the 95/95 limit, based on an acceptable correlation.

An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

The following analysis criteria are used by Westinghouse for ease in interpreting the transient results:

- Pressure in the RCS and Main Steam System should be maintained below 110% of the design values. Since a steamline rupture results in a depressurization of these systems, this is not a concern.
- Fuel cladding should be maintained by ensuring that the minimum DNBR remains above the 95/95 limit, based on an acceptable correlation. Although fuel failures are not precluded for Condition IV events, the analysis demonstrates that no fuel failures occur since the DNB criterion is met. The safety analysis DNBR limit is [ ]^{a,c} * for pressures between 500 and 1000 psia based on the W-3 DNB correlation.

* The minimum DNBR limit is the V5H fuel-specific DNB limit value corresponding to the W-3 DNB correlation. This event was evaluated for the RFA-2 with the Intermediate Fuel Mixing grids Fuel Upgrade Program. It was demonstrated that the minimum DNBR limit continues to be met, corresponding to the RFA-2 with IFMs fuel-specific W-3 DNB correlation limit.

c. Analysis Results

The Main Steamline Rupture Event bounds the accidental depressurization of the main steam system as described in the response to **RAI 15.2.2 - 3.c** in TVA letter dated November 9, 2010 (Reference 7).

(14) Inadvertent Operation of Emergency Core Cooling System during Power Operation

a. Condition II

b. Acceptance Criteria

<u>Industry/NRC acceptance criteria</u>:

Pressure in the RCS and Main Steam System should be maintained below 110% of the design values.

Fuel cladding should be maintained by ensuring that the minimum DNBR remains above the 95/95 limit, based on an acceptable correlation.

An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, fuel failure must be assumed for all rods for which the DNBR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.

The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the guidelines stated in Regulatory Guide 1.53 (Application of the Single-Failure Criterion to Nuclear Plant Protection Systems).

The criterion that the maximum pressurizer pressure reached shall be below the pressurizer safety valve opening setpoint is employed to ensure that a more serious plant condition is not generated as a result of an inadvertent operation of ECCS event. Since the pressurizer water volume continues to increase due to charging flow and residual heat generation, a manual operator action is required to isolate ECCS flow for this transient. The analysis demonstrates that termination of the injection flow at 10 minutes prevents the pressurizer safety valves from opening. This precludes possible damage to the valves which could potentially generate a more serious plant condition.

With respect to DNB, this transient is trivial, as the injection of borated water into the core actually reduces the reactor power. The FSAR analysis shows that the DNBR is never less than the initial value.

Typically, for peak pressure concerns, the expansion rate of the RCS pressure is such that the pressurizer safety valves can maintain pressure at or about 2575 psia. Thus, the 2748.5 psia analysis limit (110% of design) is rarely challenged.

c. Analysis Results

Pressurizer sprays are assumed for this analysis in order to minimize pressure for the minimum DNBR case and maximize pressurizer insurge for the pressurizer filling case. Pressurizer heaters are assumed to be inoperable for the minimum DNBR case, since this yields a higher rate of pressure decrease. The opposite is assumed for the pressurizer filling case, in which the operation of the pressurizer heaters has been found to result in an increase in the pressurizer filling rate.

PORVs are assumed as an automatic pressure control function for both the minimum DNBR, and pressurizer filling cases. For the minimum DNBR case, maintaining a low pressurizer pressure is conservative. For the pressurizer filling case, availability of the PORVs provides earlier steam relief and therefore maximizes the pressurizer insurge. However, since the pressurizer filled in the WBN analysis, the final pressurizer overfill case assumed that the PORVs are unavailable. This maximizes the pressure, which is conservative for the purpose of determining whether or not the safety valves actuate.

Minimum DNBR Case:

Nuclear power starts decreasing immediately due to boron injection, but steam flow does not decrease until the turbine throttle valve goes wide open. The mismatch between load and nuclear power causes T_{avg}, pressurizer water level, and pressurizer pressure to drop. The reactor trips on low pressurizer pressure. After the trip, pressures and temperatures slowly rise since the turbine is tripped and the reactor is producing some power due to delayed neutron fissions and decay heat. The DNBR remains above its initial value throughout the transient.

The DNBR is never less than the initial value (2.11).

Pressurizer Filling Case:

Reactor trip occurs at event initiation followed by a rapid initial cooldown of the RCS. Coolant contraction results in a short-term reduction in pressurizer pressure and water level. The combination of the RCS heatup, due to residual RCS heat generation, and ECCS injected flow causes the pressure and level transients to rapidly turn around. Pressurizer water level then increases throughout the transient. Spray flow helps to condense the pressurizer steam bubble, causing a pressurizer insurge. At 10 minutes, the ECCS injection flow is terminated via operator action and the increase in pressurizer pressure stops. At no time do the pressurizer safety valves actuate.

Although the pressurizer becomes water solid just prior to SI termination, the maximum pressure reached is below the pressurizer safety valve opening setpoint.

Condition III: Infrequent Faults are described in Section 15.3 of the Unit 2 FSAR. This section starts as follows:

"15.3 CONDITION III - INFREQUENT FAULTS

By definition Condition III occurrences are faults which may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the RCS or containment barriers. The following have been grouped into this category:

(1) Loss of reactor coolant, from small ruptured pipes or from cracks in large pipes, which actuates the ECCS.

(2) Minor secondary system pipe breaks.

(3) Inadvertent loading of a fuel assembly into an improper position.

(4) Complete loss of forced reactor coolant flow.

(5) Waste gas decay tank rupture.

(6) Single rod cluster control assembly withdrawal at full power."

***(1)* Loss of Reactor Coolant from Small Ruptured Pipes or From Cracks in Large Pipes Which Actuate the Emergency Core Cooling System**

a. Condition III Event - 1

ANS Category:

- Condition III for breaks ≥ 0.375 inch diameter hole
- Condition III for breaks < 1.0 ft^2

b. Acceptance Criteria

For cases considered, the emergency core cooling system meets the acceptance criteria as presented in 10 CFR 50.46. That is:

(1) The calculated peak fuel element cladding temperature provides margin to the limit of 2200°F, based on an Fq value of 2.50.

(2) The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of zircaloy in the reactor.

(3) The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The oxidation limit of 17% of the cladding thickness is not exceeded during or after quenching.

(4) The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

c. Analysis Results

A LOCA is defined as the loss of reactor coolant at a rate in excess of the reactor

coolant normal makeup rate from breaks or openings in the reactor coolant pressure boundary (RCPB) inside primary containment up to, and including, a break equivalent in size to the largest justified pipe rupture (or in the absence of justification, a double-ended rupture of the largest pipe) in the RCPB (ANSI/ANS-51.1-1983). The maximum break size for a small break LOCA in the WBN analyses is <1 ft^2.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the RCS through the postulated break against the charging pump makeup flow (0.375 inch diameter hole) at normal RCS pressure (i.e., 2250 psia). Should a larger break occur, depressurization of the RCS causes fluid to flow to the RCS from the pressurizer, resulting in a pressure and level decrease in the pressurizer. A reactor trip occurs when the pressurizer low pressure trip setpoint is reached. The safety injection system is actuated when the appropriate pressure setpoint is reached. The consequences of the accident are limited in two ways:

(1) Reactor trip and borated water injection complement void formation in causing rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay.

(2) Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

A spectrum of small break LOCA sizes was analyzed to determine the limiting break size in terms of the highest peak cladding temperature. The small break LOCA sizes analyzed were 2, 3, 4, 6, and 8.75 inches. The NOTRUMP computer code is used in the analysis of loss-of-coolant accidents due to small breaks in the reactor coolant system. For breaks less than 1.0 ft^2, the NOTRUMP digital computer code is employed to calculate the transient depressurization of the RCS as well as to describe the mass and enthalpy of flow through the break.

Safety injection flow rate to the RCS as a function of system pressure is an input parameter. The SIS is assumed to begin delivering full flow to the RCS 27 seconds after the generation of a safety injection signal. The 27-second delay time includes the time for diesel generator startup, loading on the 6.9 kV shutdown board, and sequential loading of the centrifugal charging and safety injection pumps onto the emergency buses, with acceleration to full speed and capability for injection. The 4-inch break was determined to be the limiting break size, with a peak cladding temperature of 1183.9°F.

The transient results for the limiting 4-inch break are presented in Unit 2 FSAR Figures 15.3-3 through 15.3-8. The depressurization transient for the 4-inch break is shown in Figure 15.3-3. The extent to which the core is uncovered is shown in Figure 15.3-4. The peak cladding temperature transient is shown in Figure 15.3-5.

The small break LOCA is bounded by the large break LOCA. Large break LOCA break sizes vary from 1 ft^2 up to and including a break equivalent in size to the largest justified pipe rupture (or in the absence of justification, a double-ended rupture of the largest pipe) in the reactor coolant pressure boundary as discussed

in Unit 2 FSAR 15.4.1.

As previously stated, the 4-inch break was determined to be the limiting break size, with a peak cladding temperature of 1183.9°F. This temperature is well below the 5,080°F melting point of uranium dioxide. Subsequently, this condition will not challenge the fuel cladding integrity resulting in minimal if any 10 CFR 100 doses at the site boundary or low population zone.

To the contrary, the design basis LOCA postulates a double-ended rupture of a reactor coolant pipe with subsequent blow-down resulting in rapid depressurization of the reactor core. An increase in fuel cladding is anticipated to the point where [1] some cladding failure may occur in the hottest regions of the reactor core, [2] result in fission products release into the reactor coolant system, and [3] ultimately enter the ECCS and primary containment. The radiological consequences of a LOCA are based on Regulatory Guide 1.4, and a total of 100% of the noble gas core inventory and 25% of the core iodine inventory are assumed to be immediately available for leakage from primary containment as shown in Unit 2 FSAR Table 15.5-6. The results of the radiological consequences of a LOCA are in Unit 2 FSAR Table 15.5.-9.

(2) Minor Secondary System Pipe Breaks

a. Condition III Event - 2

ANS Category:

- Condition III for breaks < 0.1963 ft^2
- Condition IV for double ended ruptured and breaks ≥ 0.1963 ft^2

b. Acceptance Criteria

Included in this grouping are ruptures of secondary system lines which would result in steam release rates equivalent to a 6 inch diameter break or smaller. Minor secondary system pipe breaks must be accommodated with the failure of only a small fraction of the fuel elements in the reactor. Since the results of analysis presented in Unit 2 FSAR Section 15.4.2 for a major secondary system pipe rupture also meet these criteria, separate analysis for minor secondary system pipe breaks is not required.

c. Analysis Results

The evaluation of the more probable accidental opening of a secondary system steam dump, relief or safety valve is presented in Unit 2 FSAR Section 15.2.13. These analyses are illustrative of a pipe break equivalent in size to a single valve opening. These smaller equivalent pipe break sizes are also bounded by the analysis presented in Unit 2 FSAR Section 15.4.2 for the MSLB event. The analyses presented in Section 15.4.2 demonstrate that the consequence of a minor secondary system pipe break are acceptable since a DNBR of less than the limiting value does not occur even for a more critical major secondary system pipe break.

The Environmental Consequences of a Postulated Steam Line Break are analyzed

in Unit 2 FSAR Section 15.5.4. As previously stated, the small secondary steam line break is limited to a 6.0 inch line break. A large secondary steam line break is greater than a 6.0 inch line break up to and including the largest justified pipe rupture (or in the absence of justification, a double-ended rupture of the largest pipe) in the main steam line piping. Further discussion of this analysis is provided in the Condition IV Event for Unit 2 FSAR 15.4.2, *Major Secondary System Pipe Rupture*.

(3) Inadvertent Loading of a Fuel Assembly into an Improper Position

a. Condition III Event - 3

b. Acceptance Criteria

Fuel assembly enrichment errors would be prevented by administrative procedures implemented in fabrication. In the event that a single pin or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and clad temperatures will be limited to the incorrectly loaded pin or pins. Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analyses in this section confirm that resulting power distribution effects will either be readily detected by the Power Distribution Monitoring System or will cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes.

In either of the above cases, failure of only a small fraction of the fuel rods may occur, although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the RCS or containment barriers.

c. Analysis Results

Fuel and core loading errors such as can arise from the inadvertent loading of one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes which are more peaked than those calculated with the correct enrichments. There is a 5% uncertainty margin included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. The Power Distribution Monitoring System is capable of revealing any assembly enrichment error or loading error which causes power shapes to be peaked in excess of the design value.

To reduce the probability of core loading errors, each fuel assembly is marked with

an identification number and loaded in accordance with a core loading diagram. During core loading the identification number is checked before each assembly is moved into the core. Serial numbers read during fuel movement are subsequently recorded on the loading diagram as a further check on proper placing after the loading is completed. In addition to the Power Distribution Monitoring System, thermocouples are located at the outlet of about one third of the fuel assemblies in the core. There is a high probability that these thermocouples would also indicate any abnormally high coolant enthalpy rise.

This is not a NUREG 800 event and is not bounded by another scenario.

(4) Complete Loss of Forced Reactor Coolant Flow

a. Condition III Event - 4

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps (RCPs). If the reactor is at power at the time of the accident, the immediate effect of loss of forced reactor coolant flow is a rapid increase in the reactor coolant temperature and subsequent increase in reactor coolant pressure. The flow reduction and increase in coolant temperature could eventually result in DNB and subsequent fuel damage before the peak pressures exceed the values at which the integrity of the pressure boundaries would be jeopardized unless the reactor was tripped promptly.

Normal power for the reactor coolant pumps is supplied through individual buses from a transformer connected to the generator. When generator trip occurs, the buses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to provide forced coolant flow to the core. Following a turbine trip where there are no electrical faults or a thrust bearing failure which requires tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator thus ensuring full flow for 30 seconds after the reactor trip before any transfer is made.

b. Acceptance Criteria - NUREG 800 Sections 15.3.1 - 15.3.2

The RSB acceptance criteria are based on meeting the relevant requirements of NUREG-0800, Sections 15.3.1 and 15.3.2 (1981).

c. Analysis Results

1. The most limiting set-point occurred for the CLOF Under-Frequency case.

Initial Core Average Heat Flux:	0.192455E+06 Btu/hr (ft^2)
Initial Volumetric Flow:	379,100 gpm
Core Outlet Temperature:	557.3°F
Pressurizer Pressure:	2250 psia

Time (Seconds)	Heat Flux (Fraction of Initial)	Mass Flow Rate (Fraction of Initial)
3.60	0.945	0.711

2. The calculated peak RCS pressure is 2461 psia, demonstrating that the Reactor Coolant System remains below 110% of design.

3. The DNB evaluation showed that the minimum DNBR remained above the safety limit DNBR value. No fuel damage is predicted to occur since the DNBR remained above the limit value. The calculated minimum DNBR for the two CLOF are as follows:

Case	MDNBR Limit*	MDNBR Calculated
CLOF - Under-Frequency	[ ][a,c]	1.444
CLOF - Under-Voltage	[ ][a,c]	1.465

4. An evaluation for the Unit 2 Completion Program determined that the Unit 1 analysis from just prior to the installation of the replacement steam generators remains valid for Unit 2. As such, all of the initial values assumed in the analysis and all results, including figures are based on Unit 1 analysis and are not necessarily identical to Unit 2 information but are judged to be sufficiently representative.

The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR will not decrease below the design basis limit at any time during the transient.

* Minimum DNBR limits are based on the V5H fuel type and corresponding WRB-1 DNB correlation. This event was evaluated for the RAF-2 with the Intermediate Fuel Mixing Grids Fuel Upgrade Program. It was demonstrated that the minimum DNBR limit continues to be met, corresponding to the WRB-2M DNB correlation.

(5) Waste Gas Decay Tank Rupture

a. Condition III Event - 5

b. Acceptance Criteria

The maximum amount of waste gases stored occurs after a refueling shutdown at which time the gas decay tanks store the radioactive gases stripped from the reactor coolant. The postulated accident is defined as an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a waste decay tank as a consequence of a failure of a single gas decay tank or associated piping. The consequences of this postulated accident shall not exceed regulatory

limits of 10 CFR 50, Appendix A - GDC 19 to the main control room operators and 10 CFR 100 to the general public.

c. Analysis Results

The gaseous waste processing system, as discussed in Unit 2 FSAR Section 11.3, is designed to remove fission product gases from the reactor coolant. The system consists of a closed loop with waste gas compressors, waste gas decay tanks for service at power and other waste gas decay tanks for service at shutdown and startup.

The consequences of the postulated waste gas decay tank rupture are addressed in Unit 2 FSAR 15.5.2, *Environmental Consequences of a Postulated Waste Gas Decay Tank Rupture*. The source terms used in the Waste Gas Decay Tank analyses assumes 1% defective fuel rods in the reactor core. This assumption is bounded by design basis LOCA source terms; LOCA postulates a double-ended rupture of a reactor coolant pipe with subsequent blow-down resulting in rapid depressurization of the reactor core. An increase in fuel cladding is anticipated to the point where [1] some cladding failure may occur in the hottest regions of the reactor core, [2] result in fission products release into the reactor coolant system, and [3] ultimately enter the ECCS and primary containment. The radiological consequences of a LOCA are based on Regulatory Guide 1.4, and a total of 100% of the noble gas core inventory and 25% of the core iodine inventory are assumed to be immediately available for leakage from primary containment as shown. Subsequently, the results of the analyses demonstrate that the amount of radioactivity released into the environment does not exceed the values presented in 10 CFR 100.

(6) Single Rod Cluster Control Assembly Withdrawal at Full Power

a. Condition III Event - 6

The Westinghouse criterion limits the amount of fuel damage for this event at 5%. The definition of Condition III events includes the release of radioactivity that will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. This event is consistent with Condition III definition.

b. Acceptance Criteria

Due to the very low probability of multiple failures of the rod control system or operator error necessary to cause a single rod cluster control assembly (RCCA) withdrawal, some core damage is considered as an allowable consequence. This accident is therefore classified by Westinghouse as a Condition III infrequent event as defined by American Nuclear Society Safety Criteria for Design of Stationary Pressurized Water Reactor Plants. The Westinghouse criterion on the limit on fuel damage is set at 5%.

c. Analysis Results

The current WBN design basis for the single RCCA withdrawal at full power event assumes no single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. The operator could deliberately withdraw a single RCCA in the

control bank since this feature is necessary in order to retrieve an assembly should one be accidentally dropped. In the extremely unlikely event of simultaneous electrical failures which could result in single RCCA withdrawal, rod deviation and rod control urgent failure would both be displayed on the plant annunciator, and the rod position indicators would indicate the relative positions in the assemblies in the bank. The urgent failure alarm also inhibits automatic rod withdrawal.

The Single RCCA Withdrawal at Full Power analysis for WBN is limiting at the time in life with the maximum $F_{\Delta H}$. The core power distribution results show that less than 5% of the rods are above the $F_{\Delta H}$ limit of 1.59. The DNB evaluation conservatively assumes that a fuel rod with an $F_{\Delta H}$ above the 1.59 limit is in DNB and fails. Thus, based on the core power distribution analysis results, less than 5% of the fuel rods are in DNB and the fuel failure criterion is satisfied.

Accordingly, the consequences of the doses to the main control room operator and the 10 CFR 100 doses to the public of this accident are bounded by the large break LOCA analyses.

Condition IV: Limiting Faults are described in Section 15.4 of the Unit 2 FSAR. This section starts as follows:

"15.4 <u>CONDITION IV - LIMITING FAULTS</u>

Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. They are the most drastic which must be designed against and represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR Part 100. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the emergency core cooling system (ECCS) and the containment. For the purposes of this report the following faults have been classified in this category:

(1) Major rupture of pipes containing reactor coolant up to and including double ended rupture of the largest pipe in the reactor coolant system (loss of coolant accident).

(2) Major secondary system pipe ruptures.

(3) Steam generator tube rupture.

(4) Single reactor coolant pump locked rotor.

(5) Fuel handling accident.

(6) Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)."

***(1)* Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident) (Unit 2 FSAR 15.4.1)**

a. Accident Type ANS Condition

NRC SRP Type - Loss of Reactor Coolant Inventory ANS Condition IV Event

b. Acceptance Criteria

The acceptance criteria for a large break LOCA as taken from 10 CFR 50.46 follows:

- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
- Calculated changes in core geometry shall be such that the core remains amenable to 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.

c. Analysis Results

The RCS is initially at 100% power, 2250 psia, and steady state conditions.

Depressurization of the RCS and subsequent mass release to containment cause safety injection actuation by low pressurizer pressure.

The WBN Unit 2 analysis was performed with the Westinghouse ASTRUM Evaluation Model which is based on the frozen code version WCOBRA/TRAC MOD7A, Revision 6. Analysis assumptions included a total peaking factor (F_Q) of 2.50, and enthalpy rise factor (FΔH) of 1.65, reactor power level of 100% of 3459 MWt (conservatively bounds the license thermal power of 3411 Mwt) and 10% generator tube plugging. A calculated 95th percentile at the 95-percent confidence level Peak Cladding Temperature (PCT) of 1552°F was obtained for WBN Unit 2. Additional ECCS Evaluation Model changes and other PCT assessments, when applicable, are reported per 10 CFR 50.46 reporting requirements.

WBN Unit 2 specific sensitivity studies (referred to as confirmatory cases) were completed for [ ]^{a,c}, to verify their direction of conservatism. The results of these sensitivity studies are carried forward into the uncertainty analysis.

The uncertainty analysis is based on order statistics. The ASTRUM methodology replaces the response surface technique with a statistical sampling method where the uncertainty parameters are simultaneously sampled for each case. The ASTRUM methodology has received NRC approval for referencing in licensing calculations in WCAP-16009-P-A.

Unit 2 Results of Large Break LOCA Limiting Transient

The WBN Unit 2 PCT/MLO limiting transient is a cold leg split break (effective break area = 1.8138 ft^2.

Traditionally, cold leg breaks have been limiting for large break LOCA. Analysis experience indicates that this break location most likely causes conditions that result in flow stagnation to occur in the core. Scoping studies with WCOBRA/TRAC have confirmed that the cold leg remains the limiting break location (WCAP-12945-P-A).

The large break LOCA transient can be divided into convenient time periods in which specific phenomena occur, such as various hot assembly heatup and cooldown transients. For a typical large break, the blowdown period can be divided into the Critical Heat Flux (CHF) phase, the upward core flow phase, and the downward core flow phase. These are followed by the refill, reflood, and long-term cooling periods. (The limiting case was chosen to show a conservative representation of the response to a large break LOCA.)

Critical Heat Flux (CHF) Phase

Immediately following the cold leg rupture, the break discharge rate is subcooled and high. The regions of the RCS with the highest initial temperatures (core, upper

plenum, upper head, and hot legs) begin to flash to steam, the core flow reverses and the fuel rods begin to go through departure from nucleate boiling (DNB). The fuel cladding rapidly heats up while the core power shuts down due to voiding in the core. This phase is terminated when the water in the lower plenum and downcomer begins to flash. The mixture swells and intact loop pumps, still rotating in single-phase liquid, push this two-phase mixture into the core.

Upward Core Flow Phase

Heat transfer is improved as the two-phase mixture is pushed into the core. This phase may be enhanced if the pumps are not degraded, or if the break discharge rate is low due to saturated fluid conditions at the break. If pump degradation is high or the break flow is large, the cooling effect due to upward flow may not be significant. The intact loop remains in single-phase liquid flow for several seconds, resulting in enhanced upward core flow cooling. This phase ends as the lower plenum mass is depleted, the loop flow becomes two-phase, and the pump head degrades.

Downward Core Flow Phase

The loop flow is pushed into the vessel by the intact loop pumps and decreases as the pump flow becomes two-phase. The break flow begins to dominate and pulls flow down through the core, up the downcomer to the broken loop cold leg, and out the break. While liquid and entrained liquid flow provide core cooling, the top of core vapor flow best illustrates this phase of core cooling. Once the system has depressurized to the accumulator pressure, the accumulators begin to inject cold borated water into the intact cold legs. During this period, due to steam upflow in the downcomer, a portion of the injected ECCS water is calculated to be bypassed around the downcomer and out the break. As the system pressure continues to fall, the break flow, and consequently the downward core flow, is reduced. The core begins to heat up as the system pressure approaches the containment pressure and the vessel begins to fill with ECCS water.

Refill Phase

As the refill period begins, the core begins a period of heatup and the vessel begins to fill with ECCS water. This period is characterized by a rapid increase in cladding temperatures at all elevations due to the lack of liquid and steam flow in the core region. This period continues until the lower plenum is filled and the bottom of the core begins to reflood and entrainment begins.

Early Reflood Phase

During the early reflood phase, the accumulators begin to empty and nitrogen enters the system. This forces water into the core, which then boils, causing system repressurization, and the lower core region begins to quench. During this time, core cooling may increase due to vapor generation and liquid entrainment. During the reflood period, the core flow is oscillatory as cold water periodically rewets and quenches the hot fuel cladding, which generates steam and causes system repressurization. The steam and entrained water must pass through the vessel upper plenum, the hot legs, the steam generators, and the reactor coolant pumps before it is vented out of the break. This flow path resistance is overcome by the downcomer water elevation head, which provides the gravity driven reflood force.

From the later stage of blowdown to the beginning of reflood, the accumulators rapidly discharge borated cooling water into the RCS, filling the lower plenum and contributing to the filling of the downcomer. The pumped ECCS water aids in the filling of the downcomer and subsequently supplies water to maintain a full downcomer and complete the reflood period. As the quench front progresses up the core, the PCT location moves higher into the top core region. As the vessel continues to fill, the PCT location is cooled and the early reflood period is terminated.

Late Reflood Phase

The late reflood phase is characterized by boiling in the downcomer. The mixing of ECCS water with hot water and steam from the core, in addition to the continued heat transfer from the hot vessel metal, reduces the subcooling of water in the lower plenum and downcomer.

The saturation temperature is dictated by the containment backpressure. For WBN, which has a low containment pressure after the LOCA, boiling does occur and has a significant effect on the gravity reflood. Vapor generated in the downcomer reduces the driving head which results in a reduced core reflood rate. The top core elevations experience a second reflood heatup, which exceeds the first.

Consequences of Large Break LOCA

The consequences of the large break LOCA analyses are provided in FSAR 15.5.3. The design basis LOCA postulates a double-ended rupture of a reactor coolant pipe with subsequent blow-down resulting in rapid depressurization of the reactor core. An increase in fuel cladding is anticipated to the point where [1] some cladding failure may occur in the hottest regions of the reactor core, [2] result in fission products release into the reactor coolant system, and [3] ultimately enter the ECCS and primary containment. The radiological consequences of a LOCA are based on Regulatory Guide 1.4, and a total of 100% of the noble gas core inventory and 25% of the core iodine inventory are assumed to be immediately available for leakage from primary containment as shown in attached FSAR Table 15.5-6. The results of the radiological consequences of a LOCA are in FSAR Table 15.5-9.

(2) **Major Secondary System Pipe Ruptures (Unit 2 FSAR 15.4.2.2)**

a. Accident Type

Decrease in heat removal by the secondary side ANS 18.2-73 Category

b. Acceptance Criteria

a. Pressure in the Reactor Coolant System and Main Steam System should be maintained below 110% of the design pressures.

b. Any fuel damage that may occur during the transient should be of a sufficiently limited extent so that the core will remain in place and geometrically intact with no loss of core cooling capability.

c. Any activity release must be such that the calculated doses at the site boundary are well within guidelines of 10 CFR 100.

The following Westinghouse internal criteria is used to conservatively ensure meeting the basic requirements (Reference 4.3.3.1-13.2 and 4.3.3.1-13.3):

> No bulk boiling shall occur in the primary coolant system following a feedline rupture prior to the time that the heat removal capability of the steam generators, being fed by auxiliary feedwater, exceeds the NSSS heat generation rate.

c. Analysis Results

The most limiting statepoint occurred for the hypothetical steamline break with offsite power available.

Time (sec)	Pressure (psia)	Heat Flux Fraction	Inlet Cold (°F)	Temp Hot (°F)	Flow Frac	Boron (ppm)	Reactivity (%Dr)	Density (gm/cc)
57.4	603.22	0.016	398.7	479.5	1.0	16.45	0.015	0.829

For all cases analyzed, no fuel damage occurred since the DNBR remained above the limit value. The actual minimum DNBR value is 3.93, hence, there is a large margin to the DNBR limit* for the hypothetical break with offsite power available.

* The minimum DNBR limit is the V5H fuel-specific DNB limit value corresponding to the W-3 DNB correlation. This event was evaluated for the RFA-2 with the Intermediate Fuel Mixing grids Fuel Upgrade Program. It was demonstrated that the minimum DNBR limit continues to be met, corresponding to the RFA-2 with IFMs fuel-specific W-3 DNB correlation limit.

The design-basis feedline rupture is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to maintain shell-side fluid inventory in the steam generators. Depending upon the size of the rupture and the plant operating conditions at the rupture, the event could cause either a cooldown (by excessive energy discharge out the rupture) or a heatup of the Reactor Coolant System (RCS) due to the reduction in the secondary side heat sink. The potential RCS cooldown resulting from a secondary system pipe rupture is evaluated as part of the consideration of the steamline break event. Therefore, the RCS heatup aspects are emphasized for the feedline rupture event.

A feedline rupture reduces the long-term ability of the secondary system to remove heat generated by the core from the RCS. The feedwater flow to the steam generators is reduced and since feedwater is subcooled this may cause the RCS temperatures to increase prior to reactor trip. Fluid inventory from the faulted steam generator may be discharged through the break, thereby reducing the amount of water available for decay heat removal after reactor trip. Finally, the break may be large enough and located such as to prevent the addition of any auxiliary feedwater after reactor trip to the faulted steam generator.

This analysis is conservatively performed at an initial power level corresponding to an anticipated future uprated value. This is not a NRC requirement, however, and

for WBN, the limiting feedline rupture case is initiated from a power level equivalent to 100.6% of nominal full power. Analyses have been performed at this power level assuming that offsite power is maintained throughout the transient and assuming a loss of offsite power at the time of reactor trip.

For both licensing basis cases analyzed, the results show that (1) the plant design is such that a feedline rupture presents no hazards to the integrity of the RCS or the Main Steam System pressure boundary, (2) any fuel damage is sufficiently limited so that the core remains in place and geometrically intact with no loss of core cooling capability, and (3) activity releases are such that the predicted site boundary doses are within the guidelines of 10 CFR 100.

The DNBR is not examined or calculated in these analyses.

(3) **Steam Generator Tube Rupture (Unit 2 FSAR 15.4.3)**

This was previously addressed in the response to **RAIs 15.4.3 - 5 and 15.4.3 - 6** in TVA's letter to the NRC dated November 9, 2010 (Reference 7).

(4) **Single Reactor Coolant Pump Locked Rotor (Unit 2 FSAR 15.4.4)**

a. Accident Type

ANS N18.2-1973 Category: Condition IV - Limiting Faults

b. Acceptance Criteria

Pressure in the RCS and Main Steam System (MSS) should be maintained below acceptable design limits, considering potential brittle and ductile failures.

The potential for core damage is evaluated on the basis that it is acceptable if the minimum Departure from Nucleate Boiling Ratio (DNBR) remains above the 95/95 limit, based on an acceptable correlation. If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability and that any offsite dose consequences must be within the guidelines of 10 CFR 100.

The following analysis criteria are used by Westinghouse to interpret the transient results.

- Pressure in the Reactor Coolant System is expected to be maintained below 110 percent of the design values. If the peak RCS pressure exceeds this value, the calculated RCS pressure transient must be sufficiently limited so that faulted condition stress limits are not exceeded for RCS components. (Note: The pressurization of the Main Steam System due to a locked rotor event is bounded by the Loss of Load/Turbine Trip analysis.)

- To meet the requirement of maintaining a coolable core geometry, Westinghouse analysis procedures require that the peak clad temperature for a locked rotor/shaft break analysis not exceed 2700°F for a Zirconium-steam reaction of

less than 16%.

c. **Analysis Results**

A single reactor coolant pump (RCP) locked rotor event is hypothesized based on the RCP impeller severely rubbing a stationary member. The function which provides reactor protection against this event is the low primary coolant loop flow trip.

If the reactor is at power, the immediate effect of a locked rotor is a rapid reduction in forced reactor coolant flow and an increase in the reactor coolant temperature. The sudden decrease in coolant flow and increase in reactor coolant temperature could lead to departure from nucleate boiling (DNB) and a subsequent increase in the fuel clad temperature which may lead to fuel clad melting, as well as producing a Zirconium-water reaction with the fuel clad. All of these conditions challenge the integrity of the fuel clad boundary.

The seizure of the RCP rotor during forward-loop flow and the shearing of the RCP shaft during reverse-loop flow are combined into the design-basis locked rotor analysis. This represents the most limiting combination of conditions for the RCP locked rotor and shaft break accidents.

The locked rotor analysis results demonstrate that (1) fuel clad damage is sufficiently limited such that the core remains geometrically intact with no loss of core cooling capability and (2) the plant design is such that the locked rotor event presents no hazard to the integrity of the RCPB.

1. The calculated peak RCS pressure is 2672 psia, demonstrating that the Reactor Coolant System remains below 110% of design. A generic study was performed by Westinghouse that addressed an initial pressurizer level including the pressurizer water level uncertainty and determined that, [

   ][a,c]

2. The peak clad temperature was calculated to be 1852°F, which is considerably less than the limit of 2200°F. The maximum zirconium-steam reaction at the core hot spot was calculated to be 0.36% by weight, which is considerably less than the limit of 16% by weight. The percent rods-in-DNB is calculated for each core reload and is verified to be less than the limit value (13%). Therefore, the calculated value is cycle specific.

The limiting statepoint for the locked rotor rods-in-DNB analysis is as follows.

Initial Core Average Heat Flux:	0.192455 x 10^6 Btu/hr-ft^2
Initial Volumetric Flow:	379,100 GPM
Core Inlet Temperature:	557.3°F
Pressurizer Pressure:	2250 psia

Time (seconds)	Heat Flux (Fraction of Initial)	Mass Flow Rate (Fraction of Initial)
3.30	0.962	0.593

(5) Fuel Handling Accident (Unit 2 FSAR 15.4.5)

a. Condition IV Event -5

ANS N18.2-1973 Category: Condition IV - Limiting Faults

In a Fuel Handling Accident (FHA), the cladding of each fuel rod is breached within one fuel assembly. Regulatory Guide 1.25 indicates the activity from the worst peak assembly is released and it is conservative to assume all rods will break in this accident, whereby maximizing the release. Regulatory Guide 1.83 requires that the case with the highest radioactivity should be released in the analysis. These assumptions are consistent with the definition of Condition IV occurrences which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material.

b. Acceptance Criteria

The consequences of this postulated accident shall not exceed regulatory limits of 10 CFR 50, Appendix A - GDC 19 to the main control room operators and 10 CFR 100 to the general public.

c. Analysis Results

The radiation dose results of the Regulatory Guide 1.25 FHA are provided. For a FHA inside containment, no allowance has been made for possible holdup or mixing in the primary containment or isolation of the primary containment as a result of a high radiation signal from the monitors in the ventilation systems for the case where containment penetrations are closed to the Auxiliary Building. However, the containment purge filters are credited. For a FHA inside containment when containment penetrations and/or the annulus are open to the Auxiliary Building Secondary Containment Enclosure spaces, the containment is isolated by a high radiation signal from monitors in the ventilation system and no credit is assumed for the containment purge filters. The result of a FHA inside the primary containment is well below the limits of 10 CFR 100.

As previously stated, the FHA assumes all of the fuel rods are damaged in one fuel assembly. Accordingly, these radioactive source terms are bounded by the design basis LOCA source terms as stated below.

The design basis LOCA postulates a double-ended rupture of a reactor coolant pipe with subsequent blow-down resulting in rapid depressurization of the reactor core. An increase in fuel cladding is anticipated to the point where [1] some cladding failure may occur in the hottest regions of the reactor core, [2] result in fission products release into the reactor coolant system, and [3] ultimately enter the ECCS and primary containment. The radiological consequences of a LOCA are based on Regulatory Guide 1.4 and a total of 100% of the noble gas core inventory and 25% of the core iodine inventory are assumed to be immediately available for leakage from primary containment.

(6) **Rupture of a Control Rod Drive Mechanism (Rod Cluster Control Assembly Ejection) (Unit 2 FSAR 15.4.6)**

See the response to **RAI FSAR 15.4.4 - 1** previously provided in TVA letter dated November 9, 2010 (Reference 7).

ENCLOSURE 2

Affidavit CAW-12-336 “Application for Withholding Proprietary Information from Public Disclosure”

Westinghouse Electric Company
Nuclear Services
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 720-0754
e-mail: greshaja@westinghouse.com
Proj letter: WBT-D-3734

CAW-12-3364

January 16, 2012

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Enclosure 4 of TVA RAI Response with Westinghouse Proprietary Brackets (T02 101210 001, "Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Response to Requests for Additional Information," Dec. 10, 2010)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-12-3364 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Tennessee Valley Authority.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-12-3364, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J A Gresham

J. A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 16th day of January 2012

Renee' Giampole

Notary Public

COMMONWEALTH OF PENNSYLVANIA
NOTARIAL SEAL
Renee Giampole, Notary Public
Penn Township, Westmoreland County
My Commission Expires September 25, 2013

(1) I am Manager, Regulatory Compliance, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Enclosure 4 of TVA RAI Response with Westinghouse Proprietary Brackets (T02 101210 001, "Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Response to Requests for Additional Information," Dec. 10, 2010)" (Proprietary), for submittal to the Commission, being transmitted by Tennessee Valley Authority letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with the NRC review of the Watts Bar Unit 2 license application.

This information is part of that which will enable Westinghouse to:

(a) Assist the customer in obtaining NRC review of the Watts Bar Unit 2 license.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of this information to its customers for purposes of plant specific safety analyses for licensing basis applications

(b) Its use by a competitor would improve their competitive position in the design and licensing of a similar product for the plant specific safety analyses.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculations and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

Tennessee Valley Authority

Letter for Transmittal to the NRC

The following paragraphs should be included in your letter to the NRC:

Enclosed are:

1. __ copies of "Enclosure 4 of TVA RAI Response with Westinghouse Proprietary Brackets (T02 101210 001, 'Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Response to Requests for Additional Information,' Dec. 10, 2010)" (Proprietary)

2. __ copies of "Enclosure 4 of TVA RAI Response with Westinghouse Proprietary Brackets (T02 101210 001, 'Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Response to Requests for Additional Information,' Dec. 10, 2010)" (Non-Proprietary)

Also enclosed is the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-12-3364, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 contains information proprietary to Westinghouse Electric Company LLC, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-12-3364 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.