From: Holden, Leslie E.:(GenCo-Nuc) [Leslie.Holden@exeloncorp.com]
Sent: Friday, January 31, 2014 9:23 PM
To: Wiebe, Joel
Cc: Borton, Kevin F:(GenCo-Nuc); Gullott, David M.:(GenCo-Nuc)
Subject: RE: Braidwood/Byron MUR Package for Proprietary and Factual Error Review - Part 2a
Attachments: 20140131 MUR Package for Proprietary and Factual Error Review - Part 2a LEH.docx

Joel,

We do not have any substantial comments on 2a at this time; looks good other than some minor typographical and editorial comments as noted on the attached . My Safety Analysis engineer was able to perform a quick review and he did not find anything that he immediately felt needed to be commented on, but he had to leave. He said that he would do a more detailed review and get back to me later tonight. Based on everyone else who has reviewed, I don't think that there will be anything substantial. If there is I will forward you an update to this e-mail.

Leslie

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From: Wiebe, Joel [mailto:Joel.Wiebe@nrc.gov]
Sent: Friday, January 31, 2014 1:07 PM
To: Holden, Leslie E.:(GenCo-Nuc)
Subject: Braidwood/Byron MUR Package for Proprietary and Factual Error Review - Part 2a

Leslie,

Here is the missing piece. I am calling it Part 2a of the MUR package for Proprietary and Factual Error Review.

It is 2a because it should have been after Part 2.

Joel

Hearing Identifier: NRR_PMDA
Email Number: 1101

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Options
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Reply Requested: No
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Expiration Date:
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3.1.2 Containment Systems Design (RIS 2002-03, Attachment 1, Section VI)

3.1.2.1 Regulatory Evaluation

In Appendix A to 10 CFR Part 50, GDC 4, "Environmental and dynamic effects design basis," addresses the environmental qualification of SSCs important to safety. The NRC staff reviewed the licensee's prediction of conditions in containment during postulated accidents. No regulation specifically addresses the determination of the mass and energy release into the containment following a postulated design basis accident. However, GDCs 16 and 50, address the requirements for the containment pressure resulting from a postulated design basis LOCA.

In Appendix A, GDC 16, "Containment design," specifies that the reactor containment and associated systems shall be provided to establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

In Appendix A, GDC 38, "Containment heat removal," specifies that a system to remove heat from the reactor containment be provided and that this system shall reduce rapidly, consistent with the functioning of other associated systems, the containment temperature and pressure following any LOCA.

In Appendix A, GDC 50, "Containment design basis," specifies that the reactor containment, including access openings, penetrations, and the containment heat removal system shall be designed to accommodate, without exceeding (with sufficient margin) the design leakage rate resulting from a design basis LOCA.

The regulations at 10 CFR Part 50, Appendix J, Option B, define P_a [pressure absolute] as the calculated peak containment internal pressure related to the design basis LOCA as specified in the TSs. As discussed in Section 3.1 of this SE input, the P_a values in TS Section 5.5.16, Containment Leakage Rate Testing Program, remain greater than the P_a values calculated for the uprate.

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Chapters 6.2.1 "Containment Functional Design," 6.2.1.2, "Subcompartment Analysis," 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents," and 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," of the SRP provide review guidance in the area of containment safety analysis.

3.1.2.2 Technical Evaluation

Short-Term (Subcompartment) LOCA Mass and Energy Release and Containment Analysis (RIS 2002-03, Attachment 1, Section VI.1.B)

Chapter 6.2.1.2, "Subcompartment Analysis," of the SRP, defines a subcompartment as any fully or partially enclosed volume within the primary containment that houses high energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within that volume. Sub-compartment analyses verify that the walls of a sub-compartment maintain their structural integrity following the rupture of any high energy line within the sub-compartment.

In Section II.2.14 of Attachment 5 to the licensee's submittal, letter, the licensee states that the current M&E sub-compartment releases are bounding for the MUR PU. The NRC staff concludes that this is expected and is acceptable without further review because RIS 2002-03 states that in areas (e.g., accident/transient analyses, components, systems) for which the existing analyses of record do bound plant operation at the proposed uprated power level, the staff will not conduct a detailed review. Therefore the SRP Chapter 6.2.1.2 continues to be met.

Since the current mass and energy release for the MUR PU remains bounding, the NRC staff concludes that the current predicted responses of the sub-compartments remain bounding and therefore GDCs 16 and 50 are met.

LOCA Long-Term Mass and Energy Release and Containment Response
(RIS 2002-03, Attachment 1, Section VI.1.B)

The license describes the LOCA mass and energy release and containment calculations in Section III.15 of Attachment 5 to the licensee's submittal.

The M&E evaluation model consists of the following Westinghouse computer codes: SATAN78, WREFLOOD10325, FROTH, and EPITOME. The Westinghouse COCO code was used for the containment analyses. These codes were used and found acceptable by the NRC staff in a previous power uprate for the Byron and Braidwood units.[1]

The licensee re-analyzed the LOCA long-term M&E release analyses to take into account "an identified inconsistency" in the M&E analyses and to include several input changes from the current analysis that are listed in Section III.15 of Attachment 5 to the licensee's June 23, 2011, letter. The inconsistency was identified in the EPITOME computer code. The licensee describes this inconsistency in its November 1, 2011 supplement.

Several other revisions were made to the M&E and containment analyses. These are listed in Section III.15 of Attachment 5 to the licensee's submittal. These changes either correct input or make the analyses more realistic. For example, the operating time of the containment spray system is revised to reflect the time specified in the emergency procedures.

As a result of these changes, the recalculated peak containment pressure and P_a, remain below the TS values currently in Section 5.5.16, Containment Leakage Rate Testing Program. Therefore, no change is required to the TS values of $P_{a.}$

The licensee states in Section III.15.5 of Attachment 5 to the submittal, that the long-term containment pressures for the Byron and Braidwood units are well below 50 percent of the peak value within 24 hours. This satisfies the requirement of GDC 38, Containment Heat Removal, and the guidance of SRP, Sections 6.2.1.1A and 6.2.1.3.

Since the analyses were done with acceptable methods and assumptions, the NRC staff finds the licensee's long-term containment LOCA analyses to be acceptable and GDC 38 is met.

Main Steam Line Break (MSLB) and Feedwater, Mass and Energy (M&E) Releases Inside Containment and Containment Response (RIS 2002-03, Attachment 1, Section VI.1.B)

[1] Letter from George F. Dick, USNRC, to Oliver D. Kingsley, President, Exelon Nuclear, Exelon Generation Company, LLC, Issuance of Amendments; Increase in Reactor Power, Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, May 4, 2001

The containment response to the main feedwater is not analyzed, consistent with the UFSAR since the specific enthalpy of the fluid discharged into the containment is less than that for the limiting MSLBs.

The licensee predicted M&E releases for the MSLB accident using the LOFTRAN computer code. The licensee calculated the containment response with the COCO computer code. These codes were used and found acceptable by the NRC staff in a previous power ~~power~~ uprate for the Byron and Braidwood units in a letter dated May 4, 2001 (ADAMS Accession No. ML011420274).

In response to a staff RAI concerning the modeling of the SGs, the licensee states:

> The Byron and Braidwood analysis used conservative steam generator modeling that is consistent with the NRC approved methodology found in WCAP-8822, "Mass and Energy Releases Following a Steam Line Rupture," September 1976. The Main Steam Line Break (MSLB) analyses use the LOFTRAN code (methodology found in WCAP-8822). The steam generator model is described in Section 4 of WCAP-7907-P-A, "LOFTRAN Code Description," April 1984. The user input consists of geometric parameters and the initial thermal/hydraulic conditions, including initial steam generator (SG) water mass. The important SG input parameters that impact the MSLB results are:
>
> - Initial SG water mass – this value has been set conservatively high.
> - The secondary SG water volume at which the SG tubes are assumed to start to uncover. This value has been set conservatively low to maximize the primary-to-secondary side heat transfer.
> - The quality transient of the break effluent is input by the user. It is set conservatively high to maximize the vapor release which maximizes the containment pressure.

The licensee states that small changes were made to some operating parameters which were evaluated using representative cases. The peak containment pressure and temperature cases were re-evaluated, as well as two additional hot-zero power limiting cases for each unit. The parameters chosen for the re-analyzed cases are given in Table III.16-1 of Section III.16 of Attachment 5 to the licensee's June 23, 2011, letter. The NRC staff has reviewed the analysis inputs and assumptions and finds them acceptable.

The licensee states that for the re-analyzed cases of the MSLB inside the containment the resultant maximum containment pressures are 34.6 psig and 31.4 psig, respectively, for Byron and Braidwood, Units 1 and 2. These values are less than the peak containment pressures of 39.3 psig for Unit 1, and 38.3 psig for Unit 2, for the current analyses of record. The pressure values are also less than the containment design pressure of 50 psig. The licensees states that the maximum containment air temperature for the peak case increased by 0.6 °F for Unit~~s~~ 1, and the current maximum air temperature for Unit 2 remains bounding.

Since the licensee used approved methods and conservative input assumptions for the MSLB inside containment analyses, the staff finds these analyses and results acceptable and SRP Chapter 6.2.1.4 is met.

The NRC staff had questions regarding a calculation of containment conditions used to derive the electrical equipment environmental qualification temperature and pressure profiles. In the November 1, 2011, submittal the licensee states:

> The electrical equipment environmental qualification (EQ) temperature profile is a composite curve that bounds the results from both the Main Steam Line Break Mass and Energy Release Inside Containment and the LOCA Mass and Energy Release Analyses. The electrical equipment EQ pressure profile is a curve that conservatively assumes the pressure equals the containment design pressure for the first twenty minutes of the event then corresponds to saturated ambient conditions for the remaining duration of the event.

This is conservative because the actual pressure is less than the design pressure. and therefore the NRC staff finds this acceptable and GDC 4 is met.

3.1.2.3 Conclusion

The licensee has re-performed the containment safety analyses to incorporate the correction to the EPITOME computer code used for long term containment analyses and several other revisions to the containment analysis. These re-analyses were done with acceptable methods and assumptions. Based on the above, the NRC staff determined that the containment analyses are acceptable and comply with GDCs 4, 16, 38, and 50, and the applicable SRP guidance.

3.1.3 Engineered Safety Features (ESF) Heating Ventilation and Air Conditioning (HVAC) (RIS 2002-03, Attachment 1, Section VI.1.F)

3.1.3.1 Regulatory Evaluation

The NRC's regulations and guidance specify criteria for control room habitability and post-accident fission product control and removal.

Environmental and dynamic effects design basis, GDC 4, requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents, including the effects of the release of post-accident fission products and toxic gases.

In Appendix A, GDC 19, "Control room," requires adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident.

In Appen~~i~~dix A, GDC 41, "Containment atmosphere cleanup," requires that systems to control fission products released into the reactor containment be provided to reduce the concentration and quality of fission products released to the environment following postulated accidents. Control of releases of radioactive materials to the environment, GDC 41, requires that the plant design include means to control the release of radioactive gaseous and liquid effluents for normal operation and anticipated operational occurrences (defined in 10 CFR Part 50, Appendix A).

In Appendix A, GDC 61, “Fuel storage and handling and radioactivity control” requires that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions.

In Appendix A, GDC 64, “Monitoring radioactivity releases,” requires that means shall be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and postulated accidents.

Guidance in SRP, Sections 6.4, Control Room Habitability System; 6.5.2, Containment Spray as a Fission Product Cleanup System; 9.4.1, Control Room Area Ventilation System; 9.4.2, Spent Fuel Pool Area Ventilation System; 9.4.3, Auxiliary and Radwaste Area Ventilation System; 9.4.4, Turbine Area Ventilation System; and 9.4.5, Engineered Safety Feature Ventilation System, contain specific review criteria.

3.1.3.2 Technical Evaluation

The following are ESFs ventilation systems which serve various equipment areas ~~the~~:

- Diesel-generator (DG) room ventilation system,
- Miscellaneous electrical equipment room ventilation system,
- Switchgear heat removal system, and
- Auxiliary building heating, ventilation, and air condition (HVAC) system,

The licensee states in Attachment 5, Section VI.1.F.ii of the June 23, 2011, letter, that:

> the diesel-generator room, miscellaneous electric equipment room, and switchgear room do not contain piping that is expected to see an increase in fluid temperature as a result of MUR power uprate implementation. In addition, the electrical equipment load demand and transmission loads are also not expected to increase as a result of MUR power uprate implementation. As such, the area heat loads in these rooms will not be impacted by MUR power uprate.

With regard to the auxiliary building HVAC system, the licensee states:

> The auxiliary building heat load under normal operation will not increase in most areas under MUR power uprate conditions. For those areas with no increase in heat load, there are no adverse operational or equipment effects. Heat loads in a limited number of areas did increase under MUR power uprate conditions. The heat load increase in these areas was minimal and was evaluated to be acceptable. It is noted that the ESF cubicle coolers only operate during operation of the corresponding pump. These unit coolers are actively cooled by Essential Service Water (ESW) during accident conditions. It is noted that the sump temperature under MUR power uprate conditions will not exceed the value used in the existing analyses. Therefore, the auxiliary building HVAC system is acceptable for the MUR power uprate.

The fuel handling area is also served by the auxiliary building ventilation system.

In response to an NRC staff concern, the licensee, in a letter dated November 1, 2011 further explained the Auxiliary Building heat loads as follows:

> The evaluation to support the MUR- PU (power uprate) evaluated the potential sources of heat input into the Auxiliary Building; however it was not a comprehensive room-by-room evaluation. The evaluation concluded that the Auxiliary Steam System is the only potential source of increased heat input to areas of the Auxiliary Building. The evaluation determined that there was no impact to ESF cubicles.
>
> The Auxiliary Steam System is fed by the Auxiliary Boiler and by Extraction Steam from the High Pressure Turbine. The Auxiliary Boiler is not impacted by the MUR-PU. The temperature of extraction steam system is expected to increase slightly (2.5 °F, 0.6 %) with MUR-PU conditions; therefore the Auxiliary Steam system temperature is also expected to increase slightly. The consequent increase in heat load is not considered significant. The Auxiliary Steam system provides heating for various auxiliary services (e.g., batching Boric Acid in the Auxiliary Building). Therefore any potential increases in temperature to the Auxiliary Building areas could only be impacted by the small increase in the temperature of the Auxiliary Steam. It should also be noted that the decrease in Steam Generator Blowdown temperature is expected to offset the increase in Auxiliary Steam. Therefore, minimal impact on the areas of the Auxiliary Building is expected.

With respect to the control room and auxiliary electrical equipment rooms, the licensee states in Section VI.1.F.i of Attachment 5 to the licensee's submittal, that:

> The control room and auxiliary electric equipment rooms do not contain piping that is expected to see an increase in fluid temperature as a result of MUR power uprate implementation. In addition, the electrical equipment load demand and transmission loads are also not expected to be increased as a result of MUR power uprate implementation. As such, the area heat loads will not be impacted by MUR power uprate.

Based on the above, the NRC staff concludes that the impact of the uprate on the Byron and Braidwood safety-related heating ventilation and air conditioning systems is not significant, as expected. The NRC staff therefore finds operation of these systems at MUR conditions to be acceptable.

3.1.3.3 Conclusion

Based on the above, the NRC determined that the increase in heat loads due to the uprate in the CR and on the ESF ventilation systems is not significant. The NRC staff therefore concludes that, Byron, Unit Nos.1 and 2, and Braidwood, Units 1 and 2, remain in compliance with GDCs 4, 19 41, 60, 61, and 64, and the applicable SRP.

3.2 Accident Analysis

3.2.1 Bounded Accident Analyses (RIS 2002-03, Attachment 1, Section II)

The proposed uprate~~s~~ is based on a redistribution of analytical margin originally required of ECCS evaluation models performed in accordance with the requirements set forth in 10 CFR Part 50, Appendix K, "ECCS Evaluation Models." Appendix K mandated consideration of 102 percent of the licensed power level for ECCS evaluation models of light- water reactors. The

NRC approved a change to the requirements of 10 CFR 50, Appendix K, on June 1, 2000. The change provided licensees with the option of maintaining the 2 percent power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying a reduced margin for ECCS evaluation based on the accounting of uncertainties due to instrumentation error.

In large part, the basis for acceptability of this proposed amendment is that the MUR power level conditions are bounded by the current analyses of record.

3.2.1.1 Regulatory Evaluation

Early revisions of 10 CFR 50.46, Appendix K, required licensees to base their LOCA analysis on an assumed power level of at least 102 percent of the licensed thermal power level to account for power measurement uncertainty. The NRC later modified this requirement to permit licensees to justify a smaller margin for power measurement uncertainty. Licensees may apply the reduced margin to operate the plant at a level higher than the previously licensed power. The licensee proposed to use a Cameron LEFM CheckPlus system to decrease the uncertainty in the measurement of FW flow, thereby, decreasing the power level measurement uncertainty from 2.0 percent to 0.37 percent.

The NRC staff used the guidelines in RIS 2002-03 to determine the acceptability of the proposed amendment.

3.2.1.2 Technical Evaluation

Although the licensee generally concluded that existing analyses were bounding of uprated plant operation with reduced uncertainty, the analyses were shown to be bounding in one of three different ways:

- For analyses that assume steady-state plant operation with a core power of 3672 MWt, there is a 2 percent margin for power measurement uncertainty at the CLTP, 3586.6 MWt. These analyses are bounding also of plant operation at the uprated RTP of 3645 MWt, with operating margin;
- For analyses that assume steady-state plant operation with a core power of 100 percent, the licensee evaluated accident or transient, and reanalyzed as necessary; or,
- Zero-power transients were not re-analyzed.

Table 3.2.1 – Bounded Accident and Transient Analyses

Transient/Accident	Analytic Power Level (% CLTP)	NRC Review Comments
Main Steam Line Break Mass and Energy Releases Outside Containment	102	Acceptable
Natural Circulation Cooldown	102	Acceptable

Inadvertent Opening of a Steam Generator Relief or Safety Valve	NA	Bounded by other Analysis Acceptable
Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	NA	No such regulator at the plants
Loss of Nonemergency AC Power to the Plant Auxiliaries (Loss of Offsite Power)	102	Acceptable
Loss of Normal Feedwater Flow	102	Acceptable
Feedwater System Pipe Break	102	Acceptable
Reactor Coolant Pump Shaft Seizure (Locked Rotor)/Reactor Coolant Pump Shaft Break/Locked Rotor with Loss of Offsite Power	PCT/ RCS Overpressure 102 100	Bounded Reanalyzed See 3.2.2~~3~~
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition	0	Acceptable
Rod Cluster Control Assembly Misoperation	100	Acceptable[2]
Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	NA	TS Precludes Acceptable
Chemical and Volume and Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	NA	Not Power Dependent Acceptable
Spectrum of Rod Cluster Control Assembly Ejection Accidents	102 0	Acceptable Acceptable
Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	NA	Bounded by other analysis including Inadvertent ECCS Acceptable
Failure of Small Lines Carrying Primary Coolant Outside Containment	NA	Bounded by LOCA and SBLOCA Acceptable
Loss of Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary (Best Estimate LOCA)	102	Acceptable
Small Break LOCA Analysis	102	Acceptable

[2] The NRC staff verified that the licensee used the NRC approved methodology in WCAP-11394 and concluded that analysis is not sensitive to power and therefore the existing analysis is acceptable.

Post-LOCA Long-Term Core Cooling/Subcriticality	102	Acceptable

3.2.1.3 Conclusion

RIS 2002-03~~,~~ indicates that in areas (e.g., accident/transient analyses, components, systems) for which the existing analyses of record do bound plant operation at the proposed uprated power level, the staff will not conduct a detailed review. The NRC staff therefore finds the licensee’s analyses that were performed at 102 percent of the CLTP level acceptable without detailed review. The NRC staff concludes that the licensee’s analysis performed at 0 percent of the CLTP will not change at the proposed uprated power level and therefore are acceptable. The NRC staff concludes that items in Table 3.2.1 identified as 100 percent or N/A are acceptable based on the information in the comment column.

3.2.2 Accident Analysis Not Bounded by Current Analysis of Record (AOR) (RIS 2002-03, Attachment 1, Section III)

The licensee reviewed their current analysis of record and reanalyzed the accidents that were not bounded by the proposed MUR power level.

Table 3.2.2~~3 1~~ – Accident and Transient Analyses

Transient/Accident	Analytic Power Level (% CLTP)
Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature	100
Feedwater System Malfunctions Causing an Increase in Feedwater Flow	100
Excessive Increase in Secondary Steam Flow	100
Steam System Piping Failure at Zero Power	0
Steam System Piping Failure at Full Power	100
Loss of External Load/Turbine Trip/Inadvertent Closure of Main Steam Isolation Valves/Loss of Condenser Vacuum and Other Events Causing a Turbine Trip	RCS Overpressure 102 Transient 100 MSS Overpressure NA
Partial Loss of Forced Reactor Coolant Flow	100
Complete Loss of Forced Reactor Coolant Flow	100
Reactor Coolant Pump Shaft Seizure (Locked Rotor)/Reactor Coolant Pump Shaft Break/Locked Rotor with Loss of Offsite Power	PCT/ RCS Overpressure 102 100
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	8 (limiting case) 100 60

	10
Inadvertent Operation of Emergency Core Cooling During Power Operation	102 (Peak Pressurizer Volume) 100
Inadvertent Opening of a Pressurizer Safety or Relief Valve	100
Steam Generator Tube Rupture	102
Anticipated Transients without Scram (ATWS)	100

The licensee also re-analyzed some accidents for other issues that they found and adjusted ~~them~~ for the proposed MUR power level. The licensee also proposed to adopt VIPRE sub-channel analysis code. The re-analysis using the VIPRE code used a core power level of 3648 MWt for the MUR departure from nucleate boiling (DNB) analyses. This is a 1.7 percent increase to the CLTP and is consistent with revised thermal design procedure (RTDP). The VIPRE usage, as well as, the RTDP methodology was reviewed by NRC staff. The NRC staff verified that the MUR power uprate DNBR calculations are based on a minimum measured flow of 386,000 gpm and supports the TS changes to require RCS flow to be greater than or equal to 386,000 gpm.

WCAP-14565-P-A Safety Evaluation (SE) Conditions

The NRC staff reviewed the four conditions of the SE for the staff approval of the WCAP-14565 TR for the use of the VIPRE-01 code. The staff reviewed the conditions and the licensee responses to them.

Condition 1: *Selection of the appropriate critical heat flux (CHF) correlation, DNBR [departure from nucleate boiling ratio] limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal.*

The licensee provided the limits in their response. It included the WRB-2 correlation limit of 1.17 for the VIPRE DNBR calculations for the VANTAGE+ fuel. The licensee also stated that the fuel design will not change for the MUR and therefore the fuel dependent parameters in the DNBR calculations are unchanged.

Condition 2: *Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE.*

The licensee stated that the boundary conditions are all generated by NRC-approved codes and analysis methodologies. They also stated that reactor core boundary conditions are unchanged from currently justified values besides the increase in the nominal core power of 1.7 percent.

Condition 3: *The NRC Staff's generic SER [safety evaluation report)] for VIPRE (Reference III.1-21, set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1* *and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification.*

The licensee stated the limit for the WRB-2 correlation is 1.17. The licensee also stated that the ABB-NV DNBR limit is 1.13 and the WLOP DNBR limit is 1.18 and both were previously approved for use with the VIPRE code.

Condition 4: *Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC Staff's generic review of VIPRE (Reference III.1-21) did not extend to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC Staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained.*

The licensee response stated that the MUR PU application of VIPRE replaces the THINC and FACTRAN codes and is not used in the post-CHF region.

The NRC staff reviewed the disposition of the four conditions laid out in the SE for WCAP-14565, and found that the licensee adequately addressed the conditions for the uprate. Therefore, the revision to the Safety Limit DNB correlations in TS Safety Limit 2.1.1 and the VIPRE methodology as described in WCAP-14565 dated October 1999, to TS Section 5.6.5 is acceptable.

<u>Review of Re-analyzed Events</u>

The NRC staff's review of the following accidents covered:

1. Description of the causes of the event and the description of the event itself,
2. Initial conditions,
3. Values of reactor parameters used in the analysis,
4. Analytical methods and computer codes used, and the
5. Results of the associated analysis.

The NRC staff used specific review criteria contained in Chapter 15, "Transient and Accident Analysis," of the SRP, and other guidance.

<u>Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature or an Increase in Feedwater Flow</u>

An increase in feedwater flow or reduction in temperature will result in increased subcooling in the affected SGs. The increased subcooling will then create a greater load demand on the reactor coolant system (RCS) which decreases the RCS temperature which can produce a reactivity insertion. The neutron overpower, overtemperature and overpower ΔT trips are designed to prevent power increases that could lead to DNBR becoming less than its limit value.

The NRC staff reviewed the initial conditions that the licensee used for the event. The analysis uses a 1.5 °F RCS average temperature bias and a minimum SG tube plugging level and maximum feedwater temperature.

The licensee used NRC-approved codes LOFTRAN, VIPRE-W, and ANC, as well as the RTDP to calculate DNBR. The current analysis was performed at 3600.6 MWt and the MUR PU ~~proposed~~ analysis was performed at 3672 MWt.

The licensee chose a limiting case for the four ~~plants~~ units which showed a reduction in feedwater temperature event with D5 SGs. The results showed that the resulting DNBR was greater than the safety analysis limit. The NRC staff reviewed the analysis and results and found them to be acceptable.

Excessive Increase in Secondary Steam Flow

An increase in secondary steam flow creates a mismatch between the reactor core and the SG load demand. The RPS signals that protect against this event include the low pressurizer pressure, over-temperature ΔT, and power range high neutron flux.

The licensee used NRC-approved code LOFTRAN as well as the RTDP to calculate DNBR. The current analysis was performed at 3600.6MWt and the MUR PU ~~proposed~~ analysis was performed at 3672 MWt.

The NRC staff reviewed the initial conditions for the event. The most limiting case assumed minimum reactivity feedback, automatic rod control and the Babco~~x~~ck & Wilcox International (BWI) SGs with zero tube plugging.

The analysis showed the worst-case minimum DNBR and it gave more than 20 percent safety analysis margin. The NRC staff reviewed the analysis and the results and found them to be acceptable.

Steam Supply Piping Failure at Zero Power

The steam break would result in an increase in steam flow initially which removes more energy from the RCS and causes a reduction in temperature and pressure. The cooldown can result in an insertion of positive reactivity. The most reactive rod cluster control assembly (RCCA) is assumed to be stuck in its fully withdrawn position and the possibility of returning to power exists.

The licensee used NRC-approved LOFTRAN, VIPRE, and ANC case along with the standard thermal design procedure (STDP) to calculate the minimum DNBR and peak linear heat rate (PLHR). The licensee reanalyzed the event to address revised reactivity feedback coefficients associated with MUR power level.

The NRC staff reviewed the initial conditions and assumptions for the event. The initial conditions included various conservative assumptions, as well as the assumption that the maximum break size corresponds to the size of the flow restricting nozzle in the two SG types. Protective functions available to provide protection for a steamline break included the safety injection (SI) system, overpower trips, redundant isolation of the main feedwater lines, and trip of the fast acting steamline stop valves.

The limiting case was found to be with the Unit~~s~~ 2s with a break size of 1.4 ft^2, alternating current (ac) power available and low T_{ave}. The minimum DNBR value is above the limit value of 1.18 and the maximum PLHR is below its limit value. The NRC staff noted that there was a reduction in margin to the limits which the licensee stated occurred due to power uprate reactivity coefficients creating a more severe return to power as well as the analysis being performed in a more conservative manner to bound cycle to cycle variations in future reloads. The NRC staff reviewed the analysis and the results and found them to be acceptable.

Steam System Piping Failure at Full-Power

The steam break would result in an increase in steam flow initially which removes more energy from the RCS and causes a reduction in temperature and pressure. The cooldown can result in an insertion of positive reactivity which can cause a power excursion.

The licensee used NRC-approved LOFTRAN, VIPRE, and ANC, as well as the RTDP to calculate the minimum DNBR and PLHR. The current analysis was performed at 3600.6 MWt and the ~~proposed~~ MUR PU analysis was performed at 3672 MWt.

The NRC staff reviewed the initial conditions and assumptions for the event. The initial conditions included maximum moderator reactivity feedback and least negative Doppler power feedback. The limiting break size that was found to bound all break sizes was 0.95ft^2. Protective functions that may be used to mitigate this event include the reactor trip, and SI.

Westinghouse applies the Condition II acceptance criteria such that damage to the fuel rods is precluded.

The limiting case shows that the power increases during the transient until the reactor trips on overpower ΔT. The minimum DNBR and PLHR were both found to be within the safety limits identified in the UFSAR Section 4.4. The NRC staff reviewed the analysis and the results and found them to be acceptable.

Loss of External Load/Turbine Trip/Inadvertent Closure of Main Steam Isolation Valves/Loss of Condenser Vacuum and Other Events Causing a Turbine Trip

The turbine trip event is the event found to be bounding for this analysis. For the event the turbine stop valves close very rapidly which cuts off steam flow to the turbine. The steam dumps (SDs) are initiated. The secondary temperature increases as well. The SDs and condenser normally accept the excess steam.

The licensee used the STDP for the maximum RCS and main steam system (MSS) pressure overpressure concerns. The RCS overpressure event was not reanalyzed as it is bounded by the AOR. The MSS overpressure event was analyzed based on the RCS overpressure case with automatic pressure control assumed and minimum SG tube plugging modeled. NRC-approved LOFTRAN is also used for the overpressure event.

For the DNB case, the licensee used NRC-approved LOFTRAN and the RTDP to calculate DNBR. The current analysis was performed at 3600.6 MWt and the ~~proposed~~ MUR PU analysis was performed at 3672 MWt. The DNB case assumed minimum SG tube plugging as well as the RCS flow rate corresponding to minimum measured flow of 386,000 gpm.

The NRC staff reviewed the initial conditions and assumptions for the event. One assumption is that the conditions in the reactor must cause the reactor trip (i.e., there is no reactor trip on the turbine trip). No credit is taken for SD and main feedwater flow is terminated at the time of the turbine trip and no auxiliary feedwater is credited. Manual rod control is modeled for conservatisms. No credit is taken for the SG PORVs. The MSSVs are at or greater than the TS limit of 3 percent.

The results for the MSS overpressure event showed that the overpressure case came in under the pressure limit of 1318.5 psia at about 1313.5 psia for the Unit~~s~~ 1s and 1310.6 psia for the Unit~~s~~ 2s. The minimum DNBR event showed that the minimum DNBR was above the safety limit. The NRC staff reviewed the analysis and results and found them to be acceptable.

Partial Loss of Forced Reactor Coolant Flow

The partial loss of forced reactor coolant flow would occur from the failure of an RCP. When the reactor is at power the loss of an reactor coolant pump (RCP) would result in a loss of coolant flow and a rapid increase in the coolant temperature which could lead to DNB.

The licensee used NRC-approved LOFTRAN and VIPRE codes along with the RTDP to calculate the minimum DNBR. The current analysis was performed at 3600.6 MWt and the ~~proposed~~ MUR PU analysis was performed at 3672 MWt. The analysis is performed assuming the loss of two RCPs with four loops in operation.

The NRC staff reviewed the initial conditions and assumptions for the event. The most negative Doppler-only power coefficient was modeled. The low RCS flow reactor trip is credit as being available to mitigate the event.

The results showed that the minimum DNBR was above the safety analysis limit.
The NRC staff reviewed the analysis and the results and found them to be acceptable.

Complete Loss of Forced Reactor Coolant Pump (RCP) Flow

The complete loss of forced reactor coolant flow would occur with the loss of all four RCPs and loss of the RCPs would cause immediate loss of coolant flow and a rapid increase in coolant temperature.

The licensee used NRC-approved LOFTRAN and VIPRE computer codes as well as the RTDP methodology to calculate a minimum DNBR. The current analysis was performed at 3600.6 MWt and the ~~proposed~~ MUR PU analysis was performed at 3672 MWt. Two cases were analyzed. One was the complete loss of all four RCPS and the other was the frequency decay event resulting in the complete loss of forced coolant flow.

The NRC staff reviewed the initial conditions and assumptions for the event. The most negative Doppler-only power coefficient was modeled. The RCP power Supply Undervolta~~t~~ge or Underfrequency and the low RCS flow reactor trip functions are credited to mitigate this event.

The frequency decay event was shown to be the limiting case. The reactor trips on the underfrequency trip signal after the frequency decay of 5 Hz/sec occurs for 1.2 seconds. The minimum DNBR was shown to be above the DNBR safety analysis limit. The NRC staff reviewed the analysis and the results and found them to be acceptable.

Reactor Coolant Pump Shaft Seizure (Locked Rotor)/Reactor Coolant Pump Shaft Break/Locked Rotor with Loss of Offsite Power

When an instantaneous RCP shaft seizure occurs, the flow through the loop reduces rapidly with no RCP coastdown. The coolant in the primary side heats up and expands causing an insurge into the pressurizer. Pressure suppression including sprays and the PORVs would actuate to lower pressure.

The licensee used NRC-approved LOFTRAN and VIPRE codes along with the RTDP methodology. The case that is performed as bounding is the locked rotor rods-in-DNB case. The codes and methodology are used to determine the percentage of fuel rods experiencing a DNBR. The current analysis was performed at 3600.6 MWt and the ~~proposed~~ MUR analysis was performed at 3672 MWt. The locked rotor is analyzed as one locked rotor with four loops operating and a concurrent loss of offsite power at the time of the reactor trip.
The staff reviewed the initial conditions and assumptions for this event. The most negative Doppler-only power coefficient was modeled. The low RCS flow reactor trip is credited as available to mitigate the event.

The results showed that the percentage of fuel rods exceeding the DNBR limit is less than the 2 percent fuel rod failures for the radiological dose calculations. The NRC staff reviewed the analysis and the results and found them to be acceptable.

Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at Power

This event occurs when an uncontrolled RCCA is withdrawn from the core at power. This can occur due to operator action or malfunction in the rod control system. The result is an increase in the core heat flux and an increase in the RCS temperature. The RPS is designed to terminate this event before the limits are exceeded.

The licensee used NRC-approved LOFTRAN code and the RTDP methodology to analyze this event. The ~~proposed~~ MUR PU analysis was performed at 3672 MWt. There are a variety of automatic RPS features which are designed to prevent core damage during this event and they include: power range neutron flux, positive neutron flux rate, overtemperature ΔT, overpower ΔT, high pressurizer pressure, and high pressurizer water level reactor trips. In addition to the RPS features there are rod withdrawal blocks, that would limit this event, which include high neutron flux, overpower ΔT, and overtemperature ΔT.

The NRC staff reviewed the initial conditions and assumptions for this event. Minimum and maximum reactivity feedback cases are analyzed. Reactor tips are assumed to be at their maximum values. The reactor trip assumes the highest worth RCCA is stuck fully withdrawn. Power levels of 10, 60, and 100 percent are considered.

This event is considered a Condition II event. The results were shown for the range of considered conditions. The minimum DNBR was greater than the safety analysis limit for each of the analyzed events. The NRC staff reviewed the analysis and the results and found them to be acceptable.

Inadvertent Operation of the Emergency Core Cooling System (ECCS) During Power Operation

The inadvertent operation of the ECCS at power could be caused by operator error, test sequence error, or a false electrical actuation signal. If the actuation signal occurs the suction of the charging pumps changes to the refueling water storage tank (RWST). The charging pumps then are aligned to start pumping the borated RWST water into the RCS. The accumulators and low head injection systems are not able to inject into the RCS at normal pressure. This event is analyzed to show: (1) show that there is no fuel clad damage, as indicated by the calculated minimum of DNBR, and (2) to show that the event will not escalate into a more serious event. In the first case, the reactor is not assumed to trip from the SI signal.

For the DNBR analysis, the SI signal is considered to not cause a reactor trip in the analyzed event. The reactor power will decrease due to the injection of borated water and the pressurizer pressure and water level decrease. The reactor will eventually trip by low pressurizer pressure trip or a manual trip.

The licensee used NRC-approved LOFTRAN code and the RTDP methodology to calculate the minimum DNBR. The current analysis was performed at 3600.6 MWt and the ~~proposed~~ MUR PU analysis was performed at 3672 MWt.

The NRC staff reviewed the initial conditions and assumptions for this event. The reanalysis was done for the DNB case. Some assumptions included: zero moderator temperature coefficient, low absolute value Doppler power coefficient, manual rod control, pressurizer heaters inoperable, reactor trip on low pressurizer pressure, no operator action, and no credit for the steam dump. This event is considered a Condition II event.

The results showed the power decreasing due to boron injection. The DNBR is shown to increase throughout the event. There was an expected decrease in the minimum DNBR due to the power increase from the initial conditions (MUR).

The NRC staff reviewed the analysis and results and found them to be acceptable for the DNB case of the Inadvertent Operation of the ECCS during Power Operation event.

For the analysis to demonstrate the inadvertent ECCS operation does not escalate into a more serious event the analysis was done at 102 percent of CLTP. The current licensing basis (CLB) analysis is documented in the final safety analysis report (FSAR) and dated 2002. The applicant states that this analysis continues to bound operation at the MUR power level. Since 2002, the NRC staff has issued a RIS 2005-29 regarding this event and the problems that can occur when the pressurizer is filled. The licensee's CLB analysis indicates the pressurizer is predicted to fill. The RIS states that no action or written response is required and the licensee has not updated its CLB analysis to address the concerns outlined in the RIS. The RIS states that the NRC staff will apply the guidance from the applicable standard review plans during reviews in which the accident analysis is revised (e.g., power uprates) and may have questions about how this issue has been addressed. Given that this accident analysis is performed at 102% power and therefore not revised for an MUR power uprate and the applicant has not updated the analysis in response to the RIS, the issue was determined to be outside the scope of the MUR. The staff intends to pursue this issue generically by clarifying the expectations in the RIS and is also considering plant-specific actions to address the issue. Since the concern in the RIS is that an event may escalate into a more serious event and not that the accident is not analyz~~s~~ed, the NRC staff concludes it is acceptable to address this issue outside the scope of the MUR.

Inadvertent Opening of a Pressurizer Safety or Relief Valve

The inadvertent opening of power-operated relief valve (PORV) would cause depressurization of the RCS. The analysis uses the more conservative assumption of the conditions of a pressurizer safety valve (PSV) opening because the PSVs have close to twice the steam flow rate relief capacity of the PORVs. The event initially starts with rapidly decreasing RCS pressure then the filling of the pressurizer.

The licensee used NRC-approved LOFTRAN code and the RTDP methodology to calculate the minimum DNBR. The current analysis was performed at 3600.6MWt and the ~~proposed~~ MUR PU analysis was performed at 3672 MWt. The NRC staff reviewed the initial conditions and assumptions for this event. The cases were run to find the most limiting result between the Braidwood and Byron Units 1 and 2 SGs. Maximum SG tube plugging and least negative Doppler-only power coefficient are assumed. The low pressurizer pressure and overtemperature ΔT reactor trips are credited to be available to mitigate the event. This event is considered a Condition II event.

The results showed the most limiting case being the Braidwood and Byron Unit 2 SGs with maximum tube plugging and minimum feedwater temperatures. The results showed the depressurization and the nuclear power, as well as the minimum DNBR. The reactor trip occurs on low pressurizer pressure. The minimum DNBR was greater than the DNBR safety analysis limit.

The NRC staff reviewed the analysis and the results and found them to be acceptable for all of the Braidwood and Byron Units for the DNB event of the inadvertent opening of a pressurizer safety or relief valve.

Steam Generator Tube Rupture (SGTR) Margin to Overfill (MTO) Steam Generator (SG)

The licensee performed the thermal and hydraulic analysis using LOFTTR2 program and methodology. The licensee looked at the failure of an intact SG PORV, failure of ruptured SG MSIV, and failure of ruptured SG feedwater control valve to determine the limiting single failure. The limiting single-failure was determined to be the failure of an intact SG PORV was found to be limiting due to the increased cooldown time.

The NRC staff reviewed the initial conditions and the assumptions for this event. For both units the SG MTO considered the minimum operating temperature with the minimum main feedwater temperature. They both also assumed the maximum SG tube plugging. The failure of a PORV on an intact SG was used for the limiting case. The secondary side volume available for the Braidwood and Byron Unit 1 SGs is 5122 ft^3 and the secondary side volume available for Braidwood and Byron Unit 2 SGs is 5955 ft^3. The licensee performed the analysis to show that the secondary side of the ruptured SG did not completely fill with water.

The event initiates with a tube rupture with water flowing from the primary to the secondary side of the SG. The RCS starts losing coolant and the pressurizer level and pressure are decreased. The reactor trip occurs on overtemperature ΔT trip signal. The reactor power decreases to decay heat and the turbine stop valves close. The SDs are unavailable due to assumed loss of offsite power. This also causes main feedwater flow to stop and the SG flow is provided by the auxiliary feedwater (AFW). The energy in the secondary side is released through the SG PORVS and safety valves. The pressurizer pressure signal starts the SI and flow is delivered to recover pressurizer level. The licensee assumes that the AFW flow to the ruptured SG will be isolated within nine minutes and the MSIV for the same SG is closed at 18 minutes. Three minutes of operator action time are assumed before cooldown is started. The cooldown is performed with two of the intact SGs since one of the intact SG PORVs is the assumed failure. After the cooldown termination temperature is reached the cooldown is terminated and a four-minute operator action time is assumed before RCS depressurization. The RCS depressurization is terminated when the RCS pressure is less than that of the ruptured SG and the pressurizer level was adequate. A three-minute operator action time is assumed before the SI is terminated.

The maximum ruptured SG water volume for the Braidwood and Byron Unit 1 case was shown to be 5068 ft^3. The result was, therefore, 54 ft^3 of MTO. The maximum ruptured SG water volume for the Braidwood and Byron Unit 2 case was shown to be 5685 ft^3. The result was therefore 270 ft^3 of MTO. Based on the positive value of the MTO the NRC staff determined that the MTO is acceptable.

The licensee will implement the following plant modifications to support the SG MTO assumptions:

- Installing safety-related air accumulator tanks to support AFW flow control
- Increase the capacity of the Unit 1 SG PORVs
- Modify Unit 2 SG PORVs to achieve analysis flow rates
- Install uninterruptible power supplies on two of the four SG PORVs
- Install a manual isolation valve upstream of each high head SI valve

The NRC staff reviewed the analysis and results and found them to be acceptable based on the demonstrated MTO.

Conclusion

The NRC staff has reviewed the licensee's analyses of the above events and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff further concludes that the licensee has demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptance fuel design limits (SAFDLs) and the reactor coolant pressure boundary (RCPB) pressure limits will not be exceeded as a result of these events. Therefore, the NRC staff finds the proposed uprate acceptable with respect to the events above.