Callaway Energy Center

May 08, 2015

ULNRC-06215

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

10 CFR 50.90

Ladies and Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
FACILITY OPERATING LICENSE NPF-30
REVISION OF TS 2.1.1.1 AND 5.6.5 TO REMOVE UNCERTAINTIES FROM THE DNBR SAFETY LIMIT AND ADOPT APPROVED WCAP-14565-P-A METHODOLOGY

Pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," Ameren Missouri (Union Electric Company) herewith transmits an application for amendment to Facility Operating License Number NPF-30 for the Callaway Plant.

The proposed amendment would modify Technical Specification (TS) requirements to adopt the NRC approved methodology described in WCAP-14565-P-A Addendum 2-P-A. This Topical Report describes the qualifications of the extended application of the ABB-NV and Westinghouse Low Pressure (WLOP) correlations as an alternative to the currently used W-3 correlation, in supplement to the primary Departure from Nucleate Boiling (DNB) correlation for Westinghouse Pressurized Water Reactor (PWR) fuel designs with the Westinghouse version of the VIPRE-01 (VIPRE) code.

Specifically, TS 5.6.5 would be revised to include WCAP-14565-P-A Addendum 2-P-A as an NRC approved analytical method for determining core operating limits for Callaway. In addition, as part of the proposed amendment, TS 2.1.1.1 would also be revised to provide a safety limit for the Departure from Nucleate Boiling Ratio (DNBR) that is aligned with the original intent of approved topical report WCAP-14483 and would reduce the need for potential cycle-specific license amendments resulting from a change in calculated uncertainties determined using methodologies previously approved for use at Callaway.

The appropriate TS Bases changes for the proposed revisions are included for information and reflect the proposed changes.

Attachments 1 through 4 provide the Technical Specification Page Markups, Proposed Technical Specification Bases Page Markups, Retyped Technical Specifications Pages, and Proposed FSAR Changes, respectively, in support of this amendment request. Attachments 2 and 4 are provided for information only. Final TS Bases changes will be processed under the program for updates per TS 5.5.14, "Technical Specifications Bases Control Program," at the time this amendment is implemented. Final FSAR changes will be processed under the process for FSAR updates pursuant to 10 CFR 50.71(e).

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92, "Issuance of amendment." In addition, pursuant to 10 CFR 51.22, "Criterion categorical exclusion or otherwise not requiring environmental review," Section (b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment. It should also be noted that this submittal does not contain new commitments.

The Callaway Onsite Review Committee has reviewed and approved the proposed changes and has approved the submittal of this amendment application.

This amendment request is being submitted in support of operations following the conclusion of Callaway refuel outage 21, scheduled for spring 2016. Ameren Missouri requests approval of the requested license amendment prior to reactor criticality during the restart from the upcoming Refuel 21 outage. Reactor criticality is currently scheduled for May 4, 2016. Ameren Missouri further requests that the license amendment be made effective upon NRC issuance, to be implemented within 90 days from the date of issuance.

In accordance with 10 CFR 50.91 "Notice for public comment; State consultation," Section (b)(1), a copy of this amendment application is being provided to the designated Missouri State official.

If there are any questions, please contact Mr. Tom Elwood at 314-225-1905.

I declare under penalty of perjury that the foregoing is true and correct.

Sincerely,

Scott A. M

Scott A. Maglio
Manager, Regulatory Affairs

Executed on: 5/8/2015

Enclosure: Description and Assessment of the Proposed Change

Attachments to the Enclosure:

1. Technical Specification Page Markups
2. Proposed Technical Specification Bases Page Markups
3. Retyped Technical Specification Pages
4. Proposed FSAR Changes

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DESCRIPTION AND ASSESSMENT OF THE PROPOSED CHANGE

DESCRIPTION AND ASSESSMENT OF THE PROPOSED CHANGE

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating License NPF-30 for Callaway Plant. The proposed changes would revise the Operating License to approve the application of WCAP-14565-P-A Addendum 2-P-A (Reference 1) and to simplify how the departure from nuclear boiling ratio (DNBR) safety limit is presented in the Technical Specifications (TS), in order to reduce the need for potentially unnecessary, subsequent license amendments.

2.0 DETAILED DESCRIPTION

COLR Analytical Method

TS 5.6.5.b identifies the approved Topical Reports and analytical methods used to determine the core operating limits. This section will be revised to add the following reference:

13. WCAP-14565-P-A Addendum 2-P-A, "Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications."

Corresponding Bases and FSAR markups for the TS 5.6.5.b change are provided in Attachment 2 and 4, respectively, for information only. Any necessary Core Operating Limits Report (COLR) revisions will be processed under the 10 CFR 50.59 process during Cycle 22 core re-design, when the Topical Report is first expected to be used for analysis at Callaway.

Rector Core Safety Limits

TS 2.1.1.1 will be revised to read:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.17 for the WRB-2 DNB correlation.

Corresponding Bases revisions for the TS 2.1.1.1 change are provided in Attachment 2 for information only.

3.0 TECHNICAL EVALUATION

COLR Analytical Method

The proposed change modifies TS requirements to adopt the methodology described in WCAP-14565-P-A, "Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications," Addendum 2, also known as W-3A. Specifically, TS 5.6.5 would be revised to include WCAP-14565-P-A Addendum 2-P-A as an NRC approved analytical method for determining core operating limits for Callaway. The Topical Report justifies the use of the ABB-NV correlation for the non-mixing grid region of Westinghouse Pressurized Water Reactors (PWRs) with

no change to the correlation form, its coefficients, or the currently licensed DNBR limit. The Topical Report also develops and validates the Westinghouse Low Pressure (WLOP) correlation and the proposed 95/95 DNBR limit for low pressure and low flow conditions.

Ameren Missouri has reviewed the NRC staff's final safety evaluation (SE) dated February 14, 2008 [Accession No. ML080360381] for WCAP-14565-P-A, Addendum 2-P-A (Revision 0). Section 4.0 of the NRC staff's evaluation included four limitations and conditions for application of the Topical Report. Ameren Missouri's response to each of the limitations and conditions is as follows:

1. The applicable range of the ABB-NV and WLOP correlations are presented in Table 1 and Table 2, respectively, of this SE.

   Response: For the departure from nucleate boiling (DNB) analyses conditions that were based on the ABB-NV and WLOP correlations, the results were confirmed to be within the parameter ranges of the DNB correlations as specified in Table 1 and Table 2, respectively.

   Discussion of the applicable ranges for DNB correlations at Callaway is included in FSAR Section 4.4.2.2.1. Table 1 and Table 2 of the WCAP-14565-P-A Addendum 2-P-A have been incorporated into the proposed FSAR 4.4.2.2.1 revisions included as Attachment 4 to this Enclosure.

2. The ABB-NV correlation and the WLOP correlation must use the same F_c factor for power shape correction as used in the primary DNB correlation for a specific fuel design.

   Response: For the DNB analyses conditions that were based on the ABB-NV and WLOP correlations, the F_c factor for power shape correction that was applied was the same as the power shape correction used for the WRB-2 correlation, which is the primary DNB correlation for the fuel at Callaway.

3. Selection of the appropriate DNB correlation, DNBR limit, engineering hot channel factors for enthalpy rise, and other fuel-dependent parameters will be justified for each application of each correlation on a plant specific basis.

   Response: The ABB-NV and WLOP DNB correlations are used for the analysis of the fuel when the primary DNB correlation is not applicable. The current ABB-NV and WLOP DNBR limits were approved for use with VIPRE. The 95/95 ABB-NV DNB correlation limit is 1.13 for Westinghouse PWR fuel design applications. The 95/95 WLOP DNB correlation limit is 1.18. The correlation limits used in the VIPRE DNBR calculations for Callaway are consistent with the approved values.

   There is no fuel design change associated with the Callaway implementation of ABB-NV and WLOP. The plant specific hot channel factors and other fuel dependent parameters in the DNBR calculations are unchanged from the currently approved values.

4. The ABB-NV correlation for Westinghouse PWR applications and the WLOP correlation must be used in conjunction with the Westinghouse version of the VIPRE-01 (VIPRE) code since

the correlations were justified and developed based on VIPRE and the associated VIPRE modeling specifications.

Response: The Westinghouse version of the VIPRE-01 subchannel analysis code, which has been qualified and approved with the ABB-NV and WLOP correlations, was used for Callaway DNB analyses involving the ABB-NV and WLOP correlations. This is reflected in the FSAR mark ups provided for information only in Attachment 4 to this Enclosure.

Reactor Core Safety Limits

The proposed TS 2.1.1.1 change provides a safety limit for Departure from Nucleate Boiling Ratio (DNBR) of ≥ 1.17 for the WRB-2 DNB correlation. The current Callaway TS specifies the design limit DNBR as the following:

2.1.1.1 The design limit departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.22 for transients analyzed using the revised thermal design procedure (RTDP) methodology and the WRB-2 DNB correlation. For non-RTDP transients analyzed using the standard thermal design procedure, the DNBR shall be maintained greater than or equal to the applicable DNB correlation limit (≥ 1.17 for WRB-2, ≥ 1.30 for W-3).

With this change, safety limit uncertainties applied to the primary WRB-2 correlation will now be applied in the applicable approved safety analysis methodology instead of being fixed in the TS. Removing the design uncertainties from the TS will provide a true safety limit consistent with approved WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," and reduce the need for license amendments resulting from a change in calculated uncertainties determined using methodologies previously approved for use at Callaway.

This change does not alter the use of the analytical methods used to determine core operating limits that have been reviewed and approved by the NRC. Removing analytical uncertainties from the TS would allow the use of current topical reports to refine those uncertainties without having to submit an amendment to the operating license, consistent with the intent of WCAP-14483-A to reduce the possibility of incorrect conclusions when determining if a safety limit is met.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements / Criteria

The regulatory requirements and/or guidance documents associated with this amendment application include the following:

- The regulatory basis for TS 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," is to ensure core operating limits are established in accordance with NRC approved methodologies and document those limits in the COLR.

- Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," provides guidance for the removal of cycle-specific parameter limits from the TS, since processing cycle-specific parameter limit changes was an unnecessary burden on both licensees and the NRC. The Generic Letter was intended to apply to those TS changes that were developed with NRC-approved methodologies. To support the removal of cycle-specific parameter limits, the Generic Letter recommends that cycle-specific parameter limit values be placed in a CORE OPERATING LIMITS REPORT (COLR), thereby eliminating the need for many reload license amendments. The COLR would be submitted to the NRC to allow continued trending of information even though NRC approval of these limits would not be required.

- 10 CFR 50.36(c)(5) requires that the TS include a category called "Administrative Control," that contains the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

There are no changes being proposed in this amendment application such that conformance or commitments to the regulatory requirements and/or guidance documents above would come into question. The evaluations documented herein confirm that Callaway Plant will continue to comply with all applicable regulatory requirements.

In conclusion, based on considerations discussed herein, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.2 <u>No Significant Hazards Consideration Determination</u>

Ameren Missouri has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   Overall protection system performance will remain within the bounds of the accident analyses since there are no design changes. The design of the reactor trip system (RTS) instrumentation will be unaffected, and thus, the protection system will continue to function in a manner consistent with the plant design basis. All applicable design, material, and construction standards will continue to be maintained.

The proposed changes will not affect any assumptions regarding accident initiators or precursors nor adversely alter the design assumptions, conditions, and configuration of the facility or the intended manner in which the plant is operated and maintained. The proposed changes will not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended functions to mitigate the consequences of an initiating event within the assumed acceptance limits.

The proposed changes do not physically alter safety-related systems nor affect the way in which safety-related systems perform their functions. TS 5.6.5.b continues to ensure that the analytical methods used to determine the core operating limits meet NRC reviewed and approved methodologies. TS 5.6.5.c, unchanged by this amendment application, will continue to ensure that applicable limits of the safety analyses are met.

The proposed change to TS 2.1.1.1 to specify only the true DNBR safety limit without the addition of analytical uncertainties does not alter the use of the analytical methods used to determine core operating limits that have been reviewed and approved by the NRC. Removing analytical uncertainties from the TS would allow the use of current topical reports to refine those uncertainties without having to submit an amendment to the operating license, consistent with the intent of WCAP-14483-A. Implementation of revisions to topical reports for Callaway Plant applications would still be reviewed in accordance with 10 CFR 50.59(c)(2)(viii) and, where required, receive prior NRC review and approval.

All accident analysis acceptance criteria will continue to be met with the proposed changes. The proposed changes will not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR. The applicable radiological dose acceptance criteria will continue to be met.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The ABB-NV correlation was originally developed for Combustion Engineering fuel designs, and has also been qualified and licensed for Westinghouse fuel applications for the fuel region below the first mixing vane grid where the W-3 correlation is currently applied. The WLOP correlation is developed for DNBR calculations at low pressure conditions. The W-3A correlations, which are based exclusively on DNB data from rod bundle tests, have a wider applicable range and are more accurate than the W-3 correlation, leading to increased DNB margin in the plant safety analyses. The NRC-approved ABB-NV and WLOP correlation 95/95 DNBR limits with the VIPRE-W code are 1.13 and 1.18, respectively.

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated, as the change is simply allowing the use of more accurate correlations when evaluating DNBR. The change does not involve any physical changes to the facility.

Likewise, revising TS 2.1.1.1 to present the DNBR safety limit calculated using the WRB-2 methodology, without uncertainties being applied, does not introduce any new or different failure mode from what has been previously been evaluated. The change does not involve any change to a methodology, including how uncertainties are calculated and accounted for, nor does it involve any physical change to the facility.

Collectively, and based on the above, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The ABB-NV correlation was originally developed for Combustion Engineering fuel designs, and has also been qualified and licensed for Westinghouse fuel applications for the fuel region below the first mixing vane grid where the W-3 correlation is currently applied. The WLOP correlation is developed for DNBR calculations at low pressure conditions. The W-3A correlations, which are based exclusively on DNB data from rod bundle tests, have a wider applicable range and are more accurate than the W-3 correlation, leading to increased DNB margin in the plant safety analyses. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

   The currently listed Safety Limit in TS 2.1.1.1 for DNBR of 1.22 is calculated with some uncertainties statistically combined into the 1.17 value calculated using the WRB-2 methodology. These uncertainties are combined using the RTDP methodology described in WCAP-11397-P-A. Callaway FSAR Section 4.4.1.1 discusses which uncertainties are statistically combined into the correlation limit.

   Revising TS 2.1.1.1 to present the DNBR safety limit calculated using the WRB-2 methodology, without uncertainties being applied, does not represent a change in methodology, but rather allows for changes in calculated uncertainties using methodologies previously approved for Callaway without requiring a license amendment. The proposed TS 2.1.1.1 revision does not represent a change in methodology for performing analyses.

   The proposed changes do not eliminate any surveillances or alter the frequency of surveillances required by the Technical Specifications. The nominal RTS and ESFAS trip setpoints (as well as the associated allowable values) will remain unchanged. None of the acceptance criteria for any accident analysis will be changed.

As there is no change to the source term, radiological release, or does mitigation functions assumed in the accident analysis, the proposed changes have no impact on the radiological consequences of a design basis accident.

Based on the above, the proposed changes do not involve a significant reduction in a margin of safety.

In consideration of all of the above, Ameren Missouri concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.0 ENVIRONMENTAL EVALUATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

6.0 REFERENCES

1. A. Leidich, et. al., "Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications," WCAP-14565-P-A Addendum 2-P-A (Proprietary), April 2008

2. D. S. Huegel, J. D. Andrachek, and C. E. Morgan, "Generic Methodology for Expanded Core Operating Limits Report," WCAP-14483-A, January 1999.

ATTACHMENT 1

TECHNICAL SPECIFICATION PAGE MARKUPS

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

1.17

2.1.1.1 The ~~design limit~~ departure from nucleate boiling ratio (DNBR) shall be maintained ≥ ~~1.22~~ for ~~transients analyzed using the revised thermal design procedure (RTDP) methodology and~~ the WRB-2 DNB correlation. ~~For non RTDP transients analyzed using the standard thermal design procedure, the DNBR shall be maintained greater than or equal to the applicable DNB correlation limit (≥ 1.17 for WRB 2, ≥ 1.30 for W 3).~~

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWd/MTU of burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

5.6 Reporting Requirements

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT."

5. WCAP-11397-P-A, "REVISED THERMAL DESIGN PROCEDURE."

6. WCAP-14565-P-A, "VIPRE-01 MODELING AND QUALIFICATION FOR PRESSURIZED WATER REACTOR NON-LOCA THERMAL-HYDRAULIC SAFETY ANALYSIS."

7. WCAP-10851-P-A, "IMPROVED FUEL PERFORMANCE MODELS FOR WESTINGHOUSE FUEL ROD DESIGN AND SAFETY EVALUATIONS."

8. WCAP-15063-P-A, "WESTINGHOUSE IMPROVED PERFORMANCE ANALYSIS AND DESIGN MODEL (PAD 4.0)."

9. WCAP-8745-P-A, "DESIGN BASES FOR THE THERMAL OVERPOWER DT AND THERMAL OVERTEMPERATURE DT TRIP FUNCTIONS."

10. WCAP-10965-P-A, "ANC: A WESTINGHOUSE ADVANCED NODAL COMPUTER CODE."

11. WCAP-11596-P-A, "QUALIFICATION OF THE PHOENIX-P/ANC NUCLEAR DESIGN SYSTEM FOR PRESSURIZED WATER REACTOR CORES."

12. WCAP-13524-P-A, "APOLLO: A ONE DIMENSIONAL NEUTRON DIFFUSION THEORY PROGRAM."

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

13. WCAP-14565-P-A Addendum 2-P-A, "Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications."

(continued)

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION BASES PAGE MARKUPS
(for information only)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which requires that the minimum departure from nucleate boiling ratio (DNBR) of the limiting rod during Condition I and II events is greater than or equal to the DNBR ~~design~~ limits. ~~In meeting this design basis~~, for Revised Thermal Design Procedure (RTDP) analyses, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation (WRB-2) predictions are combined statistically to obtain the overall DNBR uncertainty factor. This DNBR uncertainty factor is used to define the design limit DNBR, which corresponds to a 95% probability with 95% confidence that DNB will not occur on the limiting fuel rods during Condition I and II events. Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their normal values. The design limit DNBR values are 1.21 and 1.22 for thimble and typical cells, respectively, for VANTAGE 5 fuel. In addition, margin has been maintained by meeting safety analysis DNBR limits ~~of 1.55 and 1.59 for thimble and typical cells, respectively, for VANTAGE 5 fuel. The design limit DNBRs are considered design bases limits for fission barriers for consideration in the 10 CFR 50.59 process.~~ Reference 3 discusses ~~two non-RTDP transients analyzed with the W-3 DNBR correlation (the inadvertent opening of a steam generator relief or safety valve is no longer an analyzed event) and two non-RTDP transients analyzed with the WRB-2 DNBR correlation. The correlation limits for the W-3 and WRB-2 DNBR correlations are 1.3 and 1.17, respectively.~~

correlation

To meet this correlation limit design basis while accounting for uncertainties,

above the design limit DNBR to offset known DNBR penalties and to provide DNBR margin for operating and design flexibility.

non-RTDP transients. These transients are analyzed using the WRB-2, W-3, ABB-NV, or WLOP DNB correlation, as applicable for the specific transient. The correlation limits for WRB-2, W-3, ABB-NV, and WLOP are 1.17, 1.30, 1.13, and 1.18, respectively.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation

(continued)

BACKGROUND (CONTINUED)

temperature. Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. Reference 4 further discusses the fuel centerline temperature design basis.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Trip System (RTS) and steam generator safety valves prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the limiting hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System Allowable Values in Table 3.3.1-1, in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS flow, ΔI, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Protection for these reactor core SLs is provided by the proper operation of the steam generator safety valves and the following automatic reactor trip functions:

a. High pressurizer pressure trip;

(continued)

APPLICABLE SAFETY ANALYSES (CONTINUED)

b. Low pressurizer pressure trip;

c. Low reactor coolant system flow;

d. Overtemperature ΔT trip;

e. Overpower ΔT trip; and

f. Power Range Neutron Flux trip.

The SLs represent a design requirement for establishing the RTS Allowable Values identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The reactor core safety limits figure provided in the COLR shows the loci of points of THERMAL POWER, pressurizer pressure, and average temperature below which the calculated DNBR is not less than the design limit DNBR values, the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or the exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude the violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various RTS functions that the above criteria are satisfied during steady state operation, normal operating transients, and anticipated operational occurrences (AOOs). To ensure that the RTS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature ΔT and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core

(continued)

SAFETY LIMITS (CONTINUED)	exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RTS ensures that for variations in the THERMAL POWER, RCS pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs. Reference 4 discusses the fuel temperature design basis. Figure 15.0-1 of Reference 2 depicts the protection provided by the Overpower ΔT reactor trip function against fuel centerline melting.
APPLICABILITY	SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Allowable Values for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.
SAFETY LIMIT VIOLATIONS	The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable. The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.
REFERENCES	1. 10 CFR 50, Appendix A, GDC 10. 2. FSAR, Chapter 15. 3. FSAR Section 4.4.1.1. 4. FSAR Section 4.4.1.2.

ATTACHMENT 3

RETYPED TECHNICAL SPECIFICATION PAGES

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.17 for the WRB-2 DNB correlation.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWd/MTU of burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

5.6 Reporting Requirements

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT."

5. WCAP-11397-P-A, "REVISED THERMAL DESIGN PROCEDURE."

6. WCAP-14565-P-A, "VIPRE-01 MODELING AND QUALIFICATION FOR PRESSURIZED WATER REACTOR NON-LOCA THERMAL-HYDRAULIC SAFETY ANALYSIS."

7. WCAP-10851-P-A, "IMPROVED FUEL PERFORMANCE MODELS FOR WESTINGHOUSE FUEL ROD DESIGN AND SAFETY EVALUATIONS."

8. WCAP-15063-P-A, "WESTINGHOUSE IMPROVED PERFORMANCE ANALYSIS AND DESIGN MODEL (PAD 4.0)."

9. WCAP-8745-P-A, "DESIGN BASES FOR THE THERMAL OVERPOWER DT AND THERMAL OVERTEMPERATURE DT TRIP FUNCTIONS."

10. WCAP-10965-P-A, "ANC: A WESTINGHOUSE ADVANCED NODAL COMPUTER CODE."

11. WCAP-11596-P-A, "QUALIFICATION OF THE PHOENIX-P/ANC NUCLEAR DESIGN SYSTEM FOR PRESSURIZED WATER REACTOR CORES."

12. WCAP-13524-P-A, "APOLLO: A ONE DIMENSIONAL NEUTRON DIFFUSION THEORY PROGRAM."

13. WCAP-14565-P-A Addendum 2-P-A, "Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications."

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

ATTACHMENT 4

PROPOSED FSAR CHANGES
(for information only)

TABLE 1.6-2 (Sheet 21)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-14040-NP-A	Methodology Used to Develop Cold Overpressure Mitigation System Setpoints and RCS Heatup and Cooldown Limit Curves	Rev. 4	5.3.1.6.1		A
WCAP-15151	Westinghouse Archived Reactor Vessel Materials	12/98	5.3.1.6.1		
WCAP-12472-P-A	Addendum 1-A	Rev. 0	4.3.2.2.7	1/00	A
WCAP-15400	Analysis of Capsule X from Callaway Unit 1 Reactor Vessel Radiation Surveillance Program	6/00	5.3.4 Table 5.3-10		A
WCAP-14565-P-A Addendum 2-P-A	Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications	4/08	4.4.1.1		A

No changes to this page.

4.4 THERMAL AND HYDRAULIC DESIGN

4.4.1 DESIGN BASES

The overall objective of the thermal and hydraulic design of the reactor core is to provide adequate heat transfer which is compatible with the heat generation distribution in the core such that heat removal by the reactor coolant system or the emergency core cooling system (when applicable) assures that the following performances and safety criteria requirements are met:

a. Fuel damage (defined as penetration of the fission product barrier, i.e., the fuel rod clad) is not expected during normal operation and operational transients (Condition I) or any transient conditions arising from faults of moderate frequency (Condition II). It is not possible, however, to preclude a very small number of rod failures. These will be within the capability of the plant cleanup system and are consistent with the plant design bases.

b. The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged (see above definition) although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time.

c. The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

In order to satisfy the above criteria, the following design bases have been established for the thermal and hydraulic design of the reactor core.

Note: For purposes of description of the thermal and hydraulic design of the reactor core, specific information is presented in this section, with figures and tables provided. Though these exhibits may be based on a particular cycle, they should be regarded as typical and presented for illustration purposes only. Thermal and hydraulic design of the reactor core and verification of expected operation within acceptance criteria is completed each cycle per plant procedures and documented per the requirement of Technical Specification 5.6.5 and 5.6.6.

4.4.1.1 Departure from Nucleate Boiling Design Basis

Basis

There will be at least a 95 percent probability that DNB will not occur on the limiting fuel rods during normal operation, operational transients, and any transient conditions arising from faults of moderate frequency (Conditions I and II events), at a 95 percent confidence level.

Discussion

In this application the WRB-1 (OFA) and WRB-2 (V5/V+) correlations (Ref. 1, 2, 86) ~~are~~ were employed. OFA fuel is no longer used in Callaway reactor cores.

For those accidents which ~~still use the W-3 DNB correlation~~ are outside the range of applicability of the WRB-2 correlation, the DNBR correlation limits are ~~1.3~~ 1.13 for ABB-NV (RCCA Bank Withdrawal from Subcritical) and ~~1.45 (Steam System Piping~~ 1.18 for WLOP Failure at less than 1000 psia; see Reference 83). The RCCA Bank Withdrawal from Subcritical analysis in Section 15.4.1 uses the WRB-2 correlation, except for below the first mixing vane grid location (grid 2 on Figures 4.2-2, 4.2-2B and 4.2-2C) where the ~~W-3~~ ABB-NV correlation ~~must be~~ is used ~~(DNBR limit of 1.3). A safety analysis DNBR limit of 1.43 is used for V5/V+ fuel analyzed with the WRB-2 correlation for non-RTDP transients such as the RCCA Bank Withdrawal from Subcritical (above the first mixing vane grid) and the Startup of an Inactive Loop at an Incorrect Temperature. The safety analysis DNBR limits for non-RTDP transients analyzed with the W-3 correlation are: 1.43 RCCA Bank Withdrawal from Subcritical below the first mixing vane grid) and 1.50 (Main Steam Line Break, 500 - 1000 psia).~~

The design method employed to meet the DNB design basis is the revised thermal design procedure (RTDP), Reference 3. With RTDP methodology, uncertainties in the plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions are combined statistically to obtain the overall DNB uncertainty factor which is used to define the design limit DNBR that satisfies the DNB design criterion. Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their nominal values without uncertainties. This procedure is illustrated in Figure 2-1 of Reference 3.

The RTDP design limit DNBR values are 1.22 and 1.21 for the typical and thimble cells, respectively. To maintain DNBR margin to offset DNB penalties such as those due to fuel rod bow (paragraph 4.4.2.2.5) and the lower plenum flow anomaly (paragraph 4.4.2.2.6), the RTDP safety analyses were performed to DNBR limits higher than the design limit DNBR values. The difference between the design limit DNBRs and the safety analysis limit DNBRs results in available DNBR margin. The net DNBR margin, after consideration of all penalties, is available for operating and design flexibility.

The standard thermal design procedure (STDP) is used for those analyses where RTDP is not applicable. In the STDP method, the parameters used in the analysis are treated in a conservative way from a DNBR standpoint. The parameter uncertainties are applied directly to the plant safety analysis input values to give the lowest DNBR. The DNBR limit for STDP is the appropriate DNB correlation limit increased by sufficient margin to offset the applicable DNBR penalties.

The design DNBRs are used as the bases for Technical Specifications, and for consideration in evaluations completed in accordance with 10 CFR 50.59.

4.4.2.2 Critical Heat Flux Ratio or Departure from Nucleate Boiling Ratio and Mixing Technology

The minimum DNBRs for the rated power, design overpower, and anticipated transient conditions are given in Table 4.4-1. The minimum DNBR in the limiting flow channel will be downstream of the peak heat flux location (hot spot) due to the increased downstream enthalpy rise.

DNBRs are calculated by using the correlation and definitions described in Sections 4.4.2.2.1 and 4.4.2.2.2. The VIPRE-01 computer code (discussed in Section 4.4.4.5.1) is used to determine the flow distribution in the core and the local conditions in the hot channel for use in the DNB correlation. The use of hot channel factors is discussed in Section 4.4.4.3.1 (nuclear hot channel factors) and in Section 4.4.2.2.4 (engineering hot channel factors).

4.4.2.2.1 Departure from Nucleate Boiling Technology

The W-3 correlation, and several modifications of it, have been used in Westinghouse CHF calculations. The W-3 correlation was originally developed from single tube data (Ref. 7), but was subsequently modified to apply to the 0.422 in. O.D. rod "R" grid (Ref. 8) and "L" grid (Ref. 9) as well as the 0.374 in. O.D. (Ref. 10 and 11) rod bundle data. These modifications to the W-3 correlation have been demonstrated to be adequate for reactor rod bundle design. W-3 alternative correlations are discussed following the WRB-2 correlation information.

The WRB-1 (Ref. 1) correlation was developed based exclusively on the large bank of mixing vane grid rod bundle CHF data (over 1100 points) that Westinghouse has collected. The WRB-1 and WRB-2 correlations, based on local fluid conditions, represent the rod bundle data with better accuracy over a wider range of variables than the previous correlations. These correlations account directly for both typical and thimble cold wall cell effects, uniform and nonuniform heat flux profiles, and variations in rod heated length and in grid spacing.

The applicable range of variables (WRB-1 correlation) is:

Pressure	$1440 \le P \le 2490$ psia 10^6
Local mass velocity	$0.9 \le G_{loc}/\cancel{0^6} \le 3.7$ lb/ft^2-hr
Local quality	$-0.2 \le X_{loc} \le 0.3$
Heated length, inlet to CHF location	$L_h \le 14$ feet
Grid spacing	$13 \le g_{sp} \le 32$ inches
Equivalent hydraulic diameter	$0.37 \le d_e \le 0.60$ inches

Equivalent heated hydraulic diameter $0.46 \leq d_h \leq 0.59$ inches

Figure 4.4-2A shows measured critical heat flux plotted against predicted critical heat flux using the WRB-1 correlation.

Critical heat flux tests which model the 17 x 17 OFA have been performed with the results described in detail in Reference 5. It was concluded that the CHF characteristics of the 17 x 17 OFA/V5/V+ designs are not significantly different from those of the 17 x 17 STD design, and can be adequately described by the "R" grid form of the WRB-1 correlation. Furthermore, the new data can be incorporated into the "R" grid database. The WRB-2 correlation (Ref. 86) was developed to take credit for the V5 Intermediate Flow Mixing (IFM) grid design. Figure 4.4-2b shows measured critical heat flux plotted against critical heat flux using the WRB-2 correlation.

The applicable range of parameters for the WRB-2 correlation is as follows::

Pressure	$1440 \leq P \leq 2490$ psia $G_{loc}/$
Local Mass Velocity	$\leq$ 0.9 £ ~~G_{loc}~~/ $10^6 \leq 3.7$ lb/ft^2-h
Local Quality	$-0.1 \leq X_{loc} \leq 0.3$
Heated Length, inlet to CHF location	$L_h \leq 14$ ft
Grid Spacing	$10 \leq g_{sp} \leq 26$ in.
Equivalent Hydraulic Diameter	$0.37 \leq d_e \leq 0.51$ inches
Equivalent Heated Hydraulic Diameter	$0.46 \leq d_h \leq 0.59$ inches

INSERT A

~~The W-3 DNB correlation is applied to conditions which are outside the ranges of parameters for the WRB-1 and WRB-2 correlations.~~

4.4.2.2.2 Definition of Departure from Nucleate Boiling Ratio

The DNB heat flux ratio (DNBR) as applied to this design when all flow cell walls are heated is:

$$DNBR = \frac{q''_{DNB,N}}{q''_{loc}}$$

Replace with:

83. Leidich, A., et. al., "Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications," WCAP-14565-P-A Addendum 2-P-A, April 2008.

Boiling Flow Instabilities in a Cross-Connected Four-Parallel-Channel Upflow System," Proc of 5th International Heat Transfer Conference, Tokyo, September 3-7, 1974.

78. Kao, H. S., Morgan, C. D., and Parker, W. B., "Prediction of Flow Oscillation in Reactor Core Channel," Trans. ANS Vol. 16, pgs. 212-213, 1973.

79. Tong, L. S., "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," J. Nucl. Energy, 21, 241-248 (1967).

80. Ohtsubo, A., and Uruwashi, S., "Stagnant Fluid due to Local Flow Blockage," J. Nucl. Sci. Technol., 9, No. 7, p.p. 433-434, (1972).

81. Basmer, P., Kirsh, D. and Schultheiss, G. F., "Investigation of the Flow Pattern in the Recirculation Zone Downstream of Local Coolant Blockages in Pin Bundles," Automwirtschaft, 17, No. 8, p.p. 416-417, (1972). (In German).

82. Burke, T. M., Meyer, C. E. and Shefcheck J., "Analysis of Data from the Zion (Unit 1) THINC Verification Test," WCAP-8453-A, May, 1976.

83. ~~Letter from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse), January 31, 1989, Subject: Acceptance for Referencing of Licensing Topical Report, WCAP 9226 P (Proprietary) and WCAP 9227 (Non Proprietary), "Reactor Core Response to Excessive Secondary Steam Releases."~~

84. Miller, R. W., et. al., "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification," WCAP-10216-P-A, Rev. 1, Februaruy 1994.

85. Deleted.

86. Davidson, S. L., (Ed.), "VANTAGE 5 Fuel Assembly Reference Core Report ", WCAP-10444-P-A, September 1985 (Westinghouse Proprietary); WCAP 10444 Addendum 1-A. "Reference Core Report Vantage 5 Fuel Assembly", NRC Acceptance Letter dated 3/12/86.

87. Hill, K. W., Motley, F. E., Cadek, F. F., and Castulin, J. E., "Effects on Critical Heat Flux of Local Heat Flux Spikes on Local Flow Blockage in Pressurized Water Reactor Rod Bundles," ASME Paper 74-WA/BT-54, August 12, 1974.

88. Stewart, C. W., et al., "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," Volume 1-3 (Revision 3, August 1989), Volume 4 (April 1987), NP-2511-CCM-A, Electric Power Research Institute.

INSERT A

W-3 Alternative Correlation

The ABB-NV and WLOP, W-3 Alternative correlations, are based exclusively on DNB data from rod bundle tests. They have a wider applicable range and are more accurate than the W-3 correlation for the prediction of margin to DNB. They are used for DNBR calculations as an alternative to the W-3 correlation and supplement the primary WRB-2 DNB correlation.

The ABB-NV correlation was originally developed for fuel designs of Combustion Engineering designed Pressurized Water Reactors (PWRs) based on a linear relationship between the Critical Heat Flux (CHF) and local quality. The correlation includes the following parameters: pressure, local mass velocity, local equilibrium quality, distance from grid to CHF location, heated length from inlet to CHF location, and heated hydraulic diameter of the subchannel. Supplemental rod bundle data evaluation confirms that ABB-NV, with a 95/95 correlation limit of 1.13, is applicable to the fuel region below the first mixing vane grid for fuel designs that are compatible with Westinghouse designed PWRs (Reference 93). Figure 4.4-3a shows measured critical heat flux plotted against predicted heat flux using the ABB-NV correlation.

The applicable range of the ABB-NV correlation is:

Pressure (psia)	:	1750 to 2415
Local Mass Velocity (Mlbm/hr-ft^2)	:	0.8 to 3.16
Local Quality (fraction)	:	< 0.22
Heated Length, inlet to CHF location (in.)	:	48 (minimum) to 150
Heated Hydraulic Diameter Ratio	:	0.679 to 1.08
Grid Distance (in.)	:	7.3 to 24

The WLOP correlation is a modified ABB-NV correlation specifically developed for low pressure conditions and extended flow range to cover low pressure/low flow conditions. Modifications to ABB-NV were made based on test data from rod bundles containing non-mixing vane grids. The WLOP correlation with a 95/95 DNBR limit has also been validated with test data from rod bundles containing mixing vane grids (Reference 93). The WLOP correlation with a 95/95 DNBR limit of 1.18 has also been validated with test data from rod bundles containing mixing vane grids (Reference 93). Figure 4.4-3b shows measured critical heat flux plotted against predicted heat flux using the WLOP correlation.

The applicable range of the WLOP correlation is:

Pressure (psia)	:	185 to 1800
Local Mass Velocity (Mlbm/hr-ft^2)	:	0.23 to 3.07
Local Quality (fraction)	:	< 0.75
Heated Length, inlet to CHF location (in.)	:	48 (minimum) to 168
Heated Hydraulic Diameter Ratio	:	0.679 to 1.00
Grid Spacing Term (Reference 93)	:	27 to 115

Change Reference 83:

83. Leidich, A., et. al., "Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications," WCAP-14565-P-A Addendum 2-P-A, April 2008.

Figure 4.4-3a
Measured Versus Predicted Critical Heat Flux – ABB-NV Correlation

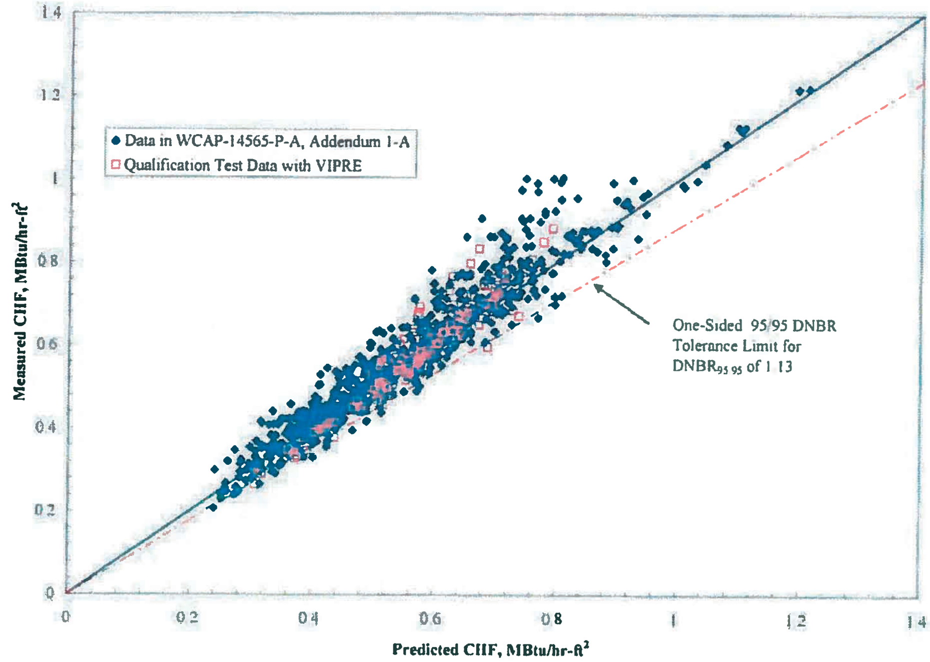
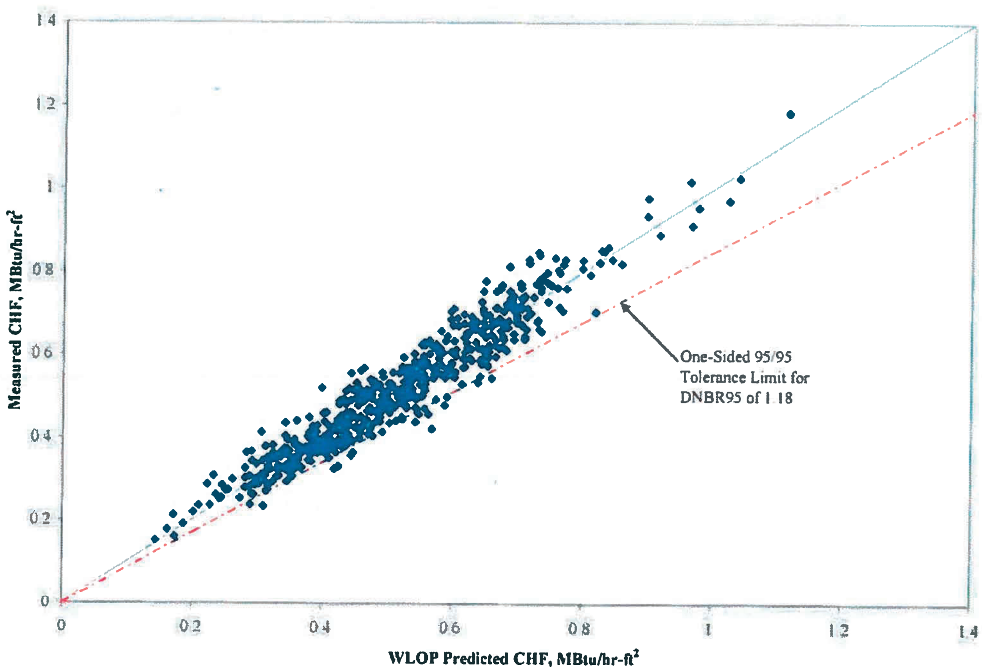


Figure 4.4-3b
Measured Versus Predicted Critical Heat Flux – WLOP Correlation



a. The NSSS rated thermal power. This power rating includes the thermal power generated by the reactor coolant pumps.

b. The rated reactor core thermal power output is 3565 MWt.

Allowances for errors in the determination of the steady-state power level are made as described in Section 15.0.3.2. The core thermal power and pump heat values used for each transient analyzed are given in Table 15.0-2.

The values of other pertinent plant parameters utilized in the accident analyses are given in Table 15.0-3.

15.0.3.2 <u>Initial Conditions</u>

For most accidents which are DNB limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the limit DNBR as described in Reference 6. The values used in the analysis of the Callaway Plant are presented in Reference 7 for the Callaway Replacement Steam Generator (RSG) Program and are contained in each cycle's Reload Safety Analysis Checklist (RSAC). The analysis power uncertainty is ± 2.0% with no bias. The rod control system (T-avg) controller channel statistical allowance (CSA) is ± 3.0°F with a bias of -1.3/+0.5°F. The pressurizer pressure control system controller CSA is ± 30 psi with an additional bias of 30 psi. These are analysis values, conservative for use but not reflective of the current plant design. This procedure is known as the revised thermal design procedure (RTDP) and the accidents analyzed with this procedure utilize the WRB-2 DNB correlation (Ref. 10). RTDP allowances may be more restrictive that non-RTDP allowances. The initial conditions for other key parameters are selected in such a manner to maximize the impact on DNBR. Minimum Measured Flow is used in all RTDP transients. This flow accounts for a flow uncertainty of ± 2.1% for calorimetric and RCS cold leg elbow tap uncertainties with a bias of +0.1% for feedwater venturi fouling.

For accidents which are not DNB limited, or for which the RTDP is not employed, the WRB-2 DNB correlation is used when coolant conditions are within the ranges of these correlations, otherwise the W-3, ABB-NV and WLOP DNB correlation ~~is~~s are used. The initial conditions are obtained by adding the maximum steady-state errors to rated values in such a manner to maximize the impact on the limiting parameter. The following conservative steady-state errors were assumed in the analysis:

a.	Core power	±2 percent allowance for calorimetric error
b.	Average reactor coolant system temperature	+4.3/-3.5°F allowance for controller deadband and measurement error
c.	Pressurizer pressure	+30/-60 psi allowance for steady-state fluctuations and measurement error

No changes.

d. Reactor coolant flow — Thermal design flow is assumed and no steady-state errors are applied

Table 15.0-2 summarizes the principal initial conditions, computer codes used, DNB correlations, and thermal hydraulic methods. Other accident specific initial conditions are given in those sections describing the accident. The level of steam generator tube plugging assumed for each transient is listed in Table 15.0-2.

15.0.3.3 Power Distribution

Out of scope.

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distributions through the placement of fuel assemblies, control rods, and operating instructions. Power distribution may be characterized by the radial factor ($F_{\Delta H}$) and the total peaking factor (F_Q). The peaking factors limits are given in the COLR.

For transients which may be DNB limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figure 15.0-1. All transients that may be DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with the design thermal power level.

For transients which may be overpower limited, the total peaking factor (F_Q) is of importance. All transients that may be overpower limited are assumed to begin with plant conditions, including power distributions which are consistent with or conservative with respect to reactor operation, as defined in the Technical Specifications.

The axial power shape discussed in Reference 9 is used in the DNBR calculation for transients analyzed at full power. It is also the limiting power shape calculated or allowed for accidents initiated at non-full power or asymmetric RCCA conditions.

The radial and axial power distributions described above are input to the VIPRE code, as described in Section 4.4.

For overpower transients that are slow with respect to the fuel rod thermal time constant, for example the chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant incident that lasts many minutes, and the excessive increase in secondary steam flow incident, which may reach equilibrium without causing a reactor trip, the fuel rod thermal evaluations are performed as discussed in Section 4.4. For overpower transients that are fast with respect to the fuel rod thermal time constant, for example, the uncontrolled RCCA bank withdrawal from subcritical or low power startup conditions and RCCA ejection incidents which result in a large power rise over a few seconds, a detailed fuel transient heat transfer calculation must be performed.

TABLE 15.0-2 SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES

Increase in feedwater flow

WLOP

	EVENT	COMPUTER CODES USED	REACTIVITY COEFFICIENTS: MODERATOR DENSITY (Δk/gm/cc)	REACTIVITY COEFFICIENTS: MODERATOR TEMPERATURE (PCM/°F)	REACTIVITY COEFFICIENTS: DOPPLER	DNB CORRELATION	REVISED THERMAL DESIGN PROCEDURE	INITIAL CORE THERMAL POWER (% RTP)
15.1	Increase in heat removal by the secondary system							
	Decrease in feedwater temperature	RETRAN	0.43	NA	Upper curve of Figure 15.0-2	WRB-2	Yes	100
	Increase in feedwater flow ~~malfunction~~ (HFP Cases)	RETRAN	0.43	NA	Upper curve of Figure 15.0-2	WRB-2	Yes	100
	(HZP Case)	RETRAN, VIPRE	Function of moderator density	NA	See Figure 15.1-14	~~W-3~~	No	0
	Excessive increase in secondary steam flow	See Section 15.1.3 for all assumptions						
	Inadvertent opening of S/G relief or safety valve	See Section 15.1.4 for all assumptions						
	Steam system piping failure	RETRAN ANC VIPRE	Function of moderator density	NA	See Figure 15.1-14	~~W-3~~	No	0 (Subcritical)
15.2	Decrease in heat removal by the secondary system							
	Loss of external electrical load and/or turbine trip DNB Case/ Pressure Case	RETRAN	NA	0	Upper curve of Figure 15.0-2	WRB-2/NA	Yes/No	100/102
	Loss of non-emergency ac to station auxiliaries	RETRAN	NA	0	Lower curve of Figure 15.0-2	NA	No	102
	Loss of normal feedwater flow	RETRAN	NA	0	Lower curve of Figure 15.0-2	NA	NA	102
	Feedwater system pipe break	RETRAN	NA	0	Lower curve of Figure 15.0-2	NA	NA	102

TABLE 15.0-2 (Sheet 2)

Increase in feedwater flow

	EVENT	REACTOR COOLANT PUMP HEAT (MWt)	REACTOR VESSEL COOLANT FLOW (gpm)	VESSEL T-AVG (°F)	PRESSURIZER PRESSURE (PSIA)	PRESSURIZER WATER LEVEL % span	FEEDWATER TEMP (°F)	EQUIVALENT S/G TUBE PLUGGING LEVEL	FULL POWER STEADY STATE FΔH	F_Q
15.1	Increase in heat removal by the secondary system									
	Decrease in feedwater temperature	14	382,630	588.4	2250	60	446	0%	NA	NA
	Increase in feedwater flow ~~malfunction~~ (HFP Cases)	14	382,630	588.4	2250	60	446	0%	NA	NA
	(HZP Case)	14	374,400	557	2250	25	100	0%	NA	NA
	Excessive increase in secondary steam flow	See Section 15.1.3 for all assumptions								
	Inadvertent opening of S/G relief or safety valve	See Section 15.1.4 for all assumptions								
	Steam system piping failure	14	374,400	557	2250	25	100	0%	NA	NA
15.2	Decrease in heat removal by the secondary system									
	Loss of external electrical load and/or turbine trip									
	DNB Case	14	382,630	588.4	2250	65	390	5%	NA	NA
	Pressure Case	14	374,400	585.4	2190	65	390	5%	NA	NA
	Loss of non-emergency ac to station auxiliaries	20	374,400	567.2	2190	43	446	0%	NA	NA
	Loss of normal feedwater flow	20	374,400	567.2	2190	43	446	0%	NA	NA
	Feedwater system pipe break	20	374,400	592.7	2190	65	446	5%	NA	NA

TABLE 15.0-2 (Sheet 3)

REACTIVITY COEFFICIENTS

	EVENT	COMPUTER CODES USED	MODERATOR DENSITY (Δk/gm/cc)	MODERATOR TEMPERATURE (PCM/°F)	DOPPLER	DNB CORRELATION	REVISED THERMAL DESIGN PROCEDURE	INITIAL CORE THERMAL POWER (%RTP)
15.3	Decrease in RCS flow rate							
	Partial/Complete loss of forced flow	RETRAN VIPRE	NA	0	Lower curve of Figure 15.0-2	WRB-2	YES	100
	Reactor coolant pump locked rotor (DNB evaluation)	RETRAN VIPRE	NA	0	Lower curve of Figure 15.0-2	WRB-2	YES	100
	Reactor coolant ~~pumo~~ pump locked rotor (peak pressure)	RETRAN VIPRE	NA	0	Lower Curve of Figure 15.0-2	NA	NO	102
15.4	Reactivity and power distribution anomalies							
	Uncontrolled RCCA bank withdrawal from subcritical	TWINKLE FACTRAN ~~THINC~~ VIPRE	See Section 15.4.1.2	See Section 15.4.1.2	See Section 15.4.1.2	WRB-2, ~~W-3~~ ABB-NV	NO	0
	Uncontrolled RCCA bank withdrawal at power	RETRAN	NA	NA/0 NA/+5 NA/+5	Upper & lower curves of Figure 15.0-2	WRB-2	YES	100 60 10
	RCCA misoperation (dropped rod)	LOFTRAN VIPRE	NA	NA	NA	WRB-2	YES	100
	Startup of an inactive loop at an incorrect temperature	See Section 15.4.4 for all assumptions						
	RCCA ejection	TWINKLE FACTRAN THINC	See Section 15.4.8.2	See Section 15.4.8.2	See Section 15.4.8.2	NA	NA	102

No changes.

15.1.5 STEAM SYSTEM PIPING FAILURE

15.1.5.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steamline would result in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is possibility that the core will become critical and return to power. A return to power following a steamline rupture is a potential problem mainly because of the high power peaking factors which exist, assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid solution delivered by the ECCS.

The analysis of a main steamline rupture is performed to demonstrate that the following criteria are satisfied:

> Assuming a stuck RCCA with or without offsite power, and assuming a single failure in the engineered safety features, the core remains in place and intact. Radiation doses do not exceed the guidelines of 10 CFR 100.
>
> Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no DNB occurs for any rupture, assuming the most reactive RCCA stuck in its fully withdrawn position. The DNBR design basis is discussed in Section 4.4.

A major steamline rupture is classified as an ANS Condition IV event. See Section 15.0.1 for a discussion of Condition IV events.

Effects of minor secondary system pipe breaks are bounded by the analysis presented in this section. Minor secondary system pipe breaks are classified as Condition III events, as described in Section 15.0.1.3.

The major rupture of a steamline is the most limiting cooldown transient, and is analyzed at zero power with no decay heat. Decay heat would retard the cooldown, thereby reducing the return to power. A detailed analysis of this transient with the most limiting break size, a double ended rupture, is presented here. The assumptions used in this analysis are discussed in Reference 3. Reference 3 also contains a discussion of the spectrum of break sizes and power levels analyzed.

During startup or shutdown evolutions when safety injection on low pressurizer pressure or low steamline pressure is blocked and steamline isolation on low steamline pressure is blocked below P-11 (pressurizer pressure less than 1970 psig), the high negative steamline pressure rate (HNPR) signal is enabled by P-11 to provide steamline isolation. A series of steamline break sensitivities in Mode 3 conditions has been

No changes.

performed using the LOFTRAN code (Ref. 1) to investigate the response of the HNPR function below P-11. Specifically, a spectrum of break sizes over a wide range of Mode 3 temperatures has been considered. The results of this study demonstrate that automatic steamline isolation is provided by the HNPR function for all but the smallest breaks for RCS temperatures from approximately the middle to the high end of the Mode 3 range. As the RCS temperatures is decreased below these values, the smaller break sizes are no longer automatically protected by the HNPR function. Finally, as the RCS temperature is reduced further, the HNPR function does not provide protection for any break size. This is consistent with the expected response of the protection function since, as the assumed RCS temperature is decreased, the initial steam generator pressure decreases as well, making it less likely that the HNPR setpoint would be reached. It should be noted that steamline isolation can also be provided by a containment pressure High-2 signal for breaks inside containment or by manual actions performed in accordance with established procedures. Furthermore, more restrictive boration requirements for conditions below P-11 make the Mode 3 steamline break scenario less limiting than the case analyzed from HZP conditions. More information on this sensitivity study can be found in Reference 9. The analyses documented in Reference 3 cover a spectrum of breaks, from HZP, partial power, and full power, including cases where the MSIVs do not close until either a containment pressure signal is generated or they are manually isolated. Reference 3, which was approved by the NRC in Reference 6, concludes that the HZP double-ended steamline break at EOL conditions discussed in Section 15.1.5.2 is a limiting case that conservatively demonstrates the compliance of Westinghouse PWRs with all applicable steamline break acceptance criteria.

The following functions provide the protection for a steamline rupture:

a. Safety injection system actuation from any of the following:

   1. Two out of three low steamline pressure signals in any one loop

   2. Two out of four low pressurizer pressure signals

   3. Two out of three high-I containment pressure signals

b. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the SIS.

c. Redundant isolation of the main feedwater lines

   Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves following reactor trip, an SIS will rapidly close all feedwater isolation valves. This signal would also trip the main feedwater pumps, close the pump discharge valves, and close the feedwater control valves. The success of this analysis is not predicated on the operation of

the main feedwater pumps nor the pump discharge valves (see Section 7.1.2.5.2 and Figure 7.2-1 sheets 13 and 14). The feedwater control valves are primary success path equipment for secondary side pipe ruptures analyzed in Section 6.2.

d. Trip of the main steam isolation valves on:

1. Safety injection system actuation derived from two out of three low steamline pressure signals in any one loop (above Permissive-11)

2. Two out of three high-2 containment pressure signals

3. Two out of three high negative steamline pressure rate signals in any one loop (used only during cooldown and heatup operations, below Permissive-11 with Tavg greater than 400°F)

Isolation valves are provided in each steamline. For breaks down-stream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would experience an uncontrolled blowdown, even if one of the isolation valves fails to close. A description of steamline isolation is included in Section 10.3. In the analysis, these valves are assumed to fully close within 17 seconds upon receipt of a steamline isolation signal following a large break in a steamline. The 17 seconds includes a 2 second signal processing delay assumption. Additionally an engineering evaluation was completed to support an increase in the main steam isolation valve stroke delay up to 60 seconds for steam generator pressures below that which corresponds to the P-11 permissive set point. This evaluation demonstrated that the acceptance criteria continue to be met for this scenario. More information on this engineering evaluation can be found in Reference 10.

Steam flow is measured by monitoring dynamic head in nozzles located in the steam generator outlet. The effective throat area of the nozzles is 1.39 square feet, which is considerably less than the main steam pipe area; thus, the nozzles also serve to limit the maximum steam flow for a break at any location.

Table 15.1-2 lists the equipment required in the recovery from a steamline rupture. Not all equipment is required for any one particular break, since the requirements will vary, depending upon postulated break size and location. Design criteria and methods of protection of safety-related equipment from the dynamic effects of postulated piping ruptures are provided in Section 3.6.

15.1.5.2 Analysis of Effects and Consequences

Method of Analysis

The analysis of the steam pipe rupture has been performed to determine:

No changes.

a. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steamline break. The RETRAN code (Ref. 8) has been used.

b. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital-computer code, VIPRE, has been used to determine if DNB occurs for the core conditions computed in item a above.

The following conditions were assumed to exist at the time of a main steamline break accident:

a. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position. Operation of the control rod banks during each burnup is restricted by the insertion limits such that addition of positive reactivity in a steamline break accident will not lead to a more adverse condition than the case analyzed.

b. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. As discussed below, this event initiates from MODE 3 (hot standby) conditions. All control and shutdown banks are on the bottom (except for the most reactive, stuck rod) since the analysis acceptance criteria are more difficult to satisfy if the shutdown banks are initially assumed to be inserted. The variation of the coefficient with temperature and pressure has been included. The k_{eff} versus temperature at 1,140 psia, corresponding to the negative moderator temperature coefficient used, is shown in Figure 15.1-11. The effect of power generation in the core on overall reactivity (due to Doppler feedback) is shown in Figure 15.1-14.

   The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculations. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high power region near the stuck rod. The stuck rod is assumed in the region of the core of lowest temperature. To verify the conservatism of this method, the reactivity, as well as the power distribution, was checked for the limiting conditions during the transient for the cases analyzed.

   This core analysis, performed with the ANC code, considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution, and nonuniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the

No changes.

effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the reactivity calculated, including the above local effects for the state points. These results verify conservatism, i.e., underprediction of negative reactivity feedback from power generation.

c. Minimum capability for injection of boric acid solution corresponding to the most restrictive single failure in the ECCS. The ECCS consists of four subsystems: 1) the passive SI accumulators, 2) the residual heat removal system, and 3) the safety injection system, and 4) the high head injection system (centrifugal charging pumps). Only the high head injection system (one CCP) and the passive SI accumulators are modeled for the steamline break accident analysis. See Table 15.1-5.

The actual modeling of the ECCS in RETRAN is described in Reference 8. No credit has been taken for the low concentration (0 ppm) borated water, which must be swept from the lines prior to the delivery of boric acid solution to the reactor coolant loops.

For the cases where offsite power is assumed, the following sequence of events occurs. After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate, and the high head safety injection pump starts. In 27 seconds (2 seconds for SIS generation (sensor) delay, 15 seconds to open RWST suction isolation valves BN-LCV-112D and E, and 10 seconds to close VCT outlet isolation valves BG-LCV-112B and C after the RWST valves are fully open), the valves are assumed to be in their final position, and the pump is assumed to be at full speed. The volume containing the low concentration (0 ppm) borated water is swept away before the boric acid solution reaches the core. This delay is included in the modeling (See Reference 7).

In cases where offsite power is not available, an additional 12-second delay is assumed to start the diesels.

d. Design value of the steam generator heat transfer coefficient, including allowance for steam generator tube fouling.

e. Since the steam generators are provided with integral flow restrictors with a 1.39-square-foot throat area, any rupture with a break area greater than 1.39 square feet, regardless of location, would have the same effect on the NSSS as the 1.39-square-foot break. The following cases have been considered in determining the core power and RCS transients:

1. Complete severance of a pipe, with the plant initially at hot standby conditions, full reactor coolant flow with offsite power available.

No changes.

2. Case (1) with loss of offsite power simultaneous with the steamline break and initiation of the safety injection signal. Loss of offsite power results in reactor coolant pump coastdown.

f. Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures are determined at end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly during the return to power phase following the steamline break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow, and, thus, are different for each case studied.

The core parameters used for each of the two cases correspond to values determined from the respective transient analysis.

Both cases above assume initial hot zero power (HZP) conditions (T_{avg} = 557°F) at time zero since this represents the worst initial condition (as discussed in Section 15.1.5.1). Should the reactor be just critical or operating at power at the time of a steamline break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip point. Following a trip at power, the RCS contains more stored energy than at HZP, the average coolant temperature is higher than at HZP, and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steamline break before the HZP conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis, which assumes no-load condition at time zero. A spectrum of steamline breaks at various power levels has been analyzed in Reference 3.

g. In computing the steam flow during a steamline break, the Moody Curve (Ref. 2) for f(L/D) = 0 is used.

Results

The calculated sequence of events for both cases analyzed is shown in Table 15.1-1.

The results presented are a conservative indication of the events which would occur, assuming a steamline rupture, since it is postulated that all of the conditions described above occur simultaneously.

Core Power and Reactor Coolant System Transient

Figures 15.1-15A through 15.1-17B show the RCS transient and core heat flux following a main steamline rupture (complete severance of a pipe) at initial HZP conditions (case a).

Offsite power is assumed to be available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs, the initiation of safety injection by low steamline pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the isolation valves in the steamlines by low steamline pressure signals, high-high containment pressure signals, or by high negative steamline pressure rate signals. Even with the failure of one valve, release is limited by main steam isolation valve closure for the other steam generators while the one generator blows down.

As shown in Figure 15.1-16B, the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) shortly before boron solution enters the RCS. The continued addition of boron results in a peak core power significantly lower than the nominal full power value.

The calculation assumes that the boric acid is mixed with, and diluted by, the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the RCS and from the centrifugal charging pump (CCP). The variation of mass flow rate in the RCS due to water density changes is included in the calculation, as is the variation of flow rate from the CCP due to changes in the RCS pressure. The CCP flow calculation includes the line losses in the system as well as the pump head curve.

Figures 15.1-18A through 15.1-20B show the salient parameters for case b, which corresponds to the case discussed above with the additional loss of offsite power at the time the safety injection signal is generated. The CCP delay time includes 12 seconds to start the diesel in addition to 27 seconds to start the centrifugal charging pump and open the valves. Criticality is achieved later, and the core power increase is slower than in the similar case with offsite power available. The ability of the emptying steam generator to extract heat from the RCS is reduced by the decreased flow in the RCS. The peak power remains well below the nominal full power value. See also Reference 7.

It should be noted that following a steamline break only one steam generator blows down completely. Thus, the remaining steam generators are still available for the dissipation of decay heat after the initial transient is over. In the case of loss of offsite power, this heat is removed to the atmosphere via the steamline atmospheric relief and safety valves.

<u>Margin to Critical Heat Flux</u>

A DNB analysis was performed for both of these cases. It was found that both cases had a minimum DNBR greater than the safety analysis limit value as discussed in Section 4.4.1.1. The ~~W-3~~ WLOP DNB~~R~~ correlation was used in this analysis~~;~~ (Reference 12). Historically, the W-3 DNB correlation had been used; see Reference 5 for the

No changes.

justification discussing the use of the W-3 correlation for low pressure applications, accepted by the NRC in Reference 6.

Out of scope.

15.1.5.3 <u>Radiological Consequences</u>

15.1.5.3.1 Method Of Analysis

15.1.5.3.1.1 Physical Model

The radiological consequences of a MSLB inside the containment are less severe than the one outside the containment because the radioactivity released will be held up inside the containment, allowing decay and plateout of the radionuclides. To evaluate the radiological consequences due to a postulated MSLB (outside the containment), it is assumed that there is a complete severance of a main steamline outside the containment.

It is also assumed that there is a simultaneous loss of offsite power, resulting in reactor coolant pump coastdown. The ECCS is actuated and the reactor trips.

The main steam isolation valves, their bypass valves, and the steamline drain valves isolate the steam generators and the main steamlines upon a signal initiated by the engineered safety features actuation system under the conditions of high negative steamline pressure rates, low steamline pressure, or high containment pressures. The main steam isolation valves are installed in the main steamlines from each steam generator downstream from the safety and atmospheric relief valves outside the containment. The break in the main steamline is assumed to occur outside of the containment. The affected steam generator (steam generator connected to a broken steamline) blows down completely. The steam is vented directly to the atmosphere.

Each of the steam generators incorporates integral flow restrictors, which are designed to limit the rate of steam blowdown from the steam generators following a rupture of the main steamline. This, in turn, reduces the cooling rate of the reactor coolant system thereby reducing the return to power.

In case of loss of offsite power, the remaining steam generators are available for dissipation of core decay heat by venting steam to the atmosphere via the atmospheric relief and safety valves. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently so that the RHR system can be utilized to cool the reactor.

15.1.5.3.1.2 Assumptions and Conditions

The major assumptions and parameters assumed in the analysis are itemized in Tables 15.1-3 and 15A-1.

15.1.5.3.3.2 Dose to Receptor at the Exclusion Area Boundary and Low-Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of a postulated MSLB have been conservatively analyzed, using asusmptions and models described. The total-body gamma doses due to immersion from direct radiation and the thyroid dose due to inhalation have been analyzed for the 0-2 hour dose at the exclusion area boundary and for the duration of the accident (0 to 8 hrs) at the low-population zone outer boundary. The results are listed in Table 15.1-4. The resultant doses are well within the guideline values of 10 CFR 100.

Out of scope.

15.1.5.4 <u>Conclusions</u>

No changes.

The analysis has shown that the criteria stated earlier in Section 15.1.5.1 are satisfied.

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable and not precluded by the criteria, the above analysis shows that the DNB design basis is met for any rupture, assuming the most reactive RCCA stuck in its fully withdrawn position.

A safety evaluation was performed to determine the impact of a potential increase in the stroke time of the feedwater isolation valves beyond the value assumed in the analyses (15 seconds) due to the installation of new valve actuators. It was concluded that the results presented in this section for the zero power steamline break event are not adversely affected by this plant modification. As such, the reported results and conclusions remain valid.

15.1.5.5 <u>Steam Line Break with Coincident Control Rod Withdrawal</u>

Out of scope.

This accident is no longer applicable to Callaway since automatic rod withdrawal is no longer available.

15.1.5.6 <u>Steam System Piping Failure at Full Power</u>

15.1.5.6.1 Identification of Causes and Accident Description

A Steamline Rupture - Full Power Core Response transient is defined as a "break" that results in an increase in steam flow from one or more steam generators. A Steamline Rupture can result from:

- An inadvertent opening of a steam generator dump, safety or relief valve
- A rupture of the main steam piping

Increased steam flow from the steam generators causes an increase in the heat extraction rate from the reactor coolant system, resulting in a reduction of primary

e. Reactor Trip

Reactor Trip is initiated on one of the following signals, depending on the break size:

- Safety Injection following a Low Steam Pressure in any steam line, or
- OPΔT in any two loops.

f. Turbine Trip

Turbine trip is initiated following reactor trip.

15.1.5.6.2.3 Results

The transient response for the Steamline Rupture - Full Power Core Response analysis is shown in Figures 15.1-21 through 15.1-26. Table 15.1-1a gives the time sequence of events for the limiting break size of 0.87 ft^2.

15.1.5.6.3 Conclusions

The analysis shows that the acceptance criteria stated above are satisfied with the improved setpoints proposed as a result of the Callaway OTΔT and OPΔT Margin Recovery Program. The DNBR safety analysis limit is met, and there is no fuel centerline melt.

Out of scope.

15.1.6 REFERENCES

No changes.

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3. Hollingsworth, S. D. and Wood, D. C., "Reactor Core Response to Excessive Secondary Steam Releases," WCAP-9226, Revision 1, (Proprietary), February, 1988, and WCAP-9227, Revision 1, (Non-Proprietary), February 1998.

4. WCAP-11397-P-A (Proprietary), WCAP-11397-A (Non-proprietary), "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1989.

5. Union Electric Letter to NRC, ULNRC-1258, dated 2-18-86.

6. Letter from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse), January 31, 1989, Subject: "Acceptance for Referencing of Licensing Topical Report, WCAP-9226-P (Proprietary) and WCAP-9227 (Non-Proprietary), Reactor Core Response to Excessive Secondary Steam Releases."

7. Union Electric Letter to NRC, ULNRC-1493, dated 4-16-87, approved by OL Amendment No. 22 dated May 4, 1987.

8. WCAP-14882-P-A, Revision 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.

9. SCP-94-162, "Callaway (SCP) Mode 3 SLB Evaluation," January 4, 1995.

10. SCP-07-19, "Main Steam Isolation Valve (MSIV) Stroke Time Evaluation Phase 2 Report Revision 0," February 16, 2007.

11. WCAP-14565-P-A (Proprietary), WCAP-15306-A (Non-Proprietary), "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. X. Sung, October 1999.

12. WCAP-14565-P-A Addendum 2-P-A (Proprietary), "Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications," A. Leidich, et al., April 2008.

TABLE 15.1-1 (Sheet 2)

Accident	Event	Time (sec)
	Feedwater isolation valves close automatically	106.7
Excessive increase in secondary steam flow		
1. Manual reactor control (minimum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached*	100
2. Manual reactor control (maximum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached*	50
3. Automatic reactor control (minimum moderator feedback)	10-percent step load increase	0.0
	Equilibrium conditions reached*	150
4. Automatic reactor control (maximum moderator feedback)	10-percent step load increase	0.0
Out of scope.	Equilibrium conditions reached*	50
Steam system piping failure		
1. Case 1 (offsite power available)	Steamline ruptures	0
	Low steamline pressure setpoint reached in faulted loop	0.510
No changes.	Low steamline pressure setpoint reached in intact loops	2.059
	Steamline isolation occurs	17.510
	Criticality attained	~23
	SI actuation	28
	Boron reaches core	100
2. Case 2 (concurrent loss of offsite power)	Steamline ruptures	0
	Low steamline pressure setpoint reached in faulted loop	0.510
	Low steamline pressure setpoint reached in intact loops	2.059
	Steamline isolation occurs	17.510
	Criticality attained	~30
	SI actuation	40

No changes.

TABLE 15.1-1 (Sheet 3)

Accident	Event	Time (sec)
	Boron reaches core	112

* Approximate time only