Comments and Proprietary Markings for NRC Draft Safety Evaluation Report for WCAP-17524, Revision 1, "AP1000 Core Reference Report" (Non-Proprietary)

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DRAFT SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS
WCAP-17524-P, REVISION 1, "AP1000 Core Reference Report"

1.0 Introduction

By letter dated March 6, 2012, Westinghouse Electric Company (Westinghouse) submitted licensee topical report (LTR) WCAP-17524-P, "AP1000 Core Reference Report" (Reference 1) to the staff of the U.S. Nuclear Regulatory Commission (NRC staff) for acceptance review (Reference 16) and approval. Westinghouse submitted the following supplemental information:

- LTR-NRC-12-56, "Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report' To Address the Impact of Thermal Conductivity Degradation on Additional Events," (Reference 2)
- LTR-NRC-12-46, "Supplemental Information to WCAP-17524, 'AP1000 Core Reference Report' To Address Thermal Conductivity Degradation," (Reference 10)
- LTR-NRC-13-26, "Supplemental Information on End-of-Life Seismic/LOCA calculations for the AP1000 Pressurized Water Reactor," (Reference 3)
- LTR-NRC-14-20, "Supplemental Information on the Impacts of Errors in the Loss of Coolant Accident Evaluation Models," (Reference 41).

By letter dated March 21, 2014, Westinghouse submitted WCAP-17524-P, Revision 1, "AP1000 Core Reference Report," (Reference 37) to capture updates to the topical report as a result of the review process.

This safety evaluation report (SER) is based on the submitted letter, supplemental information letters, and responses to requests for additional information (RAIs). Reference 37 is designed to be referenced as part of a post combined operating license (COL) licensing amendment request and in initial COL applications for licensees and applicants desiring to build the AP1000 plant design, but with the latest fuel design and core design improvements as compared with the previously approved designs found in the AP1000 standard plant design. The subject topical report (Reference 37) describes changes to the fuel system design, which relate to the following review areas of the standard review plan (SRP), NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Reference 4):

- Chapter 4.2, "Fuel System Design"
- Chapter 4.3, "Nuclear Design"
- Chapter 4.4, "Thermal and Hydraulic Design"

- Chapter 15, "Transient and Accident Analysis"

Reference 37 describes an enhanced AP1000 fuel and core design that relies on newer components currently in operational use within the existing U.S. pressurized-water reactor (PWR) fleet, a new tungsten gray rod control assembly (GRCA) design based on the approved topical report WCAP-16943-P-A (Reference 7), and a new core loading plan. This SER does not cover any plant designs beyond the AP1000 standard plant.

This SER is divided into seven sections. Section 2 presents a summary of applicable regulatory criteria and guidance, Section 3 contains a summary of the information presented in the topical report, and Section 4 contains the technical evaluation of the major components of Reference 37. These components include the following:

(1) fuel system design

(2) nuclear design

(3) thermal hydraulic design

(4) loss-of-coolant accident (LOCA)

(5) non-LOCA

Section 5 presents the conclusions of this review, and Section 6 contains the restrictions and limitations on the use of the core reference report (CRR) topical report.

2.0 Regulatory Criteria

The AP1000 CRR includes modifications to the AP1000 standard plant design that covers a variety of technical review areas. The following sections present the relevant requirements and guidance that guided the staff's review.

2.1. REQUIREMENTS

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 10, "Reactor Design," requires, in part, that control and protection systems be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation, including the effects of anticipated operational occurrences. The SAFDLs associated with the AP1000 plant design are defined by the AP1000 standard plant design (approved as design control document (DCD) Revision 19). The fuel design changes provided in Reference 37 do not affect the previously reviewed and approved SAFDLs.

Regulations in 10 CFR Part 50, Appendix A, GDC 27, "Combined Reactivity Control Systems Capability," require the following:

> The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Regulations in 10 CFR Part 50, Appendix A, GDC 35, "Emergency Core Cooling," require the following:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

2.2. RELEVANT GUIDANCE

NUREG-0800 (Reference 4) provides detailed review guidance that the staff finds acceptable in meeting the applicable regulatory requirements. In particular, NUREG-0800 sections that contain guidance relevant to this review are:

1. Section 4.2, "Fuel System Design"
2. Section 4.3, "Nuclear Design"
3. Section 4.4, "Thermal and Hydraulic Design"
4. Section 15, "Transient and Accident Analysis"

3.0 Summary of Technical Information

Reference 37 provides a detailed description and justification for a new fuel design and advanced first core (AFC) loading pattern to make use of improvements in fuel assembly design, core components and analysis methods when compared with the approved revised

AP1000 design certification (DCD Revision 19). The purpose of the CRR topical report is to provide the latest updates to be incorporated into the initial COL applications and in COL license amendment requests, as appropriate. The design changes described in Reference 37 include:

- implementation of the AFC loading pattern
- incorporation of the enhanced GRCA
- minor modifications to the rod cluster control assembly (RCCA)
- incorporation of robust protective grid (RPG)
- modifications to the grid strap heights of the mid grids and intermediate flow mixing (IFM) grids

In addition to the DCD Chapter 4 design changes listed in the CRR, the applicant re-analyzed DCD Chapter 15 accident analyses based on the new fuel assembly design. In some cases (e.g., reactivity initiated accident analysis), the methodology used in the analysis is based on an updated and previously approved methodology.

3.1. FUEL SYSTEM DESIGN

The application (Reference 37) presents evaluations and analyses based on the physical changes to the grid designs in the new AP1000 PWR fuel assembly.

Modifications to the mid and IFM grid designs are analyzed for their impacts on the applicable safety analyses. These changes are addressed in the Thermal Hydraulic Design and Accident Analysis Sections.

The RCCA design has been modified to simplify manufacturability and improve reliability. Specific changes include the use of lower carbon SS-304L, integral design of vanes and fingers with the spider body, chrome plating for wear resistance, and physical changes to reduce the chances of pellet-cladding interaction caused by irradiation swelling.

The previously approved GRCA design is replaced with a tungsten-based design, as previously reviewed and approved in WCAP-16943-P-A (Reference 7). However, similar design changes, as seen for the RCCAs, are included in the GRCA design.

The fuel assembly protective grid design has been changed to RPG, which is identical to the design used in some operating plants. The changes associated with the RPG (page 2-2 of the application (Reference 37)), are designed to mitigate vibration and fatigue induced failures, as well as primary water stress corrosion cracking. The applicant analyzes the effects of the

Comment [c1]: Editorial correction. Missing parenthesis.

protective grid design change to the GSI-191 in-vessel downstream effects analysis and concludes that no adverse impacts will be caused by the change.

3.2. NUCLEAR DESIGN

Section 3 and Appendix A Section 4.3 of the CRR presents changes and updates to the nuclear design as compared with Revision 19 of the AP1000 DCD. In Reference 1, Westinghouse proposed changes to the following areas related to the AP1000 nuclear design:

(1) the AFC, or first cycle fuel assembly loading pattern

(2) GRCA design

(3) the discrete burnable absorber design (specified to use wet annular burnable absorber (WABA) rather than Pyrex)

The nuclear design parameters affected include the minimum shutdown margin (SDM) and the enthalpy rise hot channel factor ($F_{\Delta H}$).

The revised nuclear design presented in the application (Reference 37) requires changes to the technical specifications (TS) and bases. This includes the TS limiting condition for operation (LCO) that describes the axial flux difference limits. This information is contained in Section 6.2 of the application (Reference 37).

3.3. THERMAL HYDRAULIC DESIGN

Section 4 and Appendix A of Section 4.4 of the application (Reference 37) address proposed changes to the thermal hydraulic design described in Revision 19 of the AP1000 DCD. As described in Reference 37, Westinghouse proposed changes to the following areas related to the AP1000 thermal hydraulic design:

- replace the DNBR limits,
- add two CHF correlations for use in certain conditions,
- adjust the new CHF correlations,
- apply the enthalpy rise hot channel factor ($F_{\Delta H}$),
- include the additional departure for nucleate boiling (DNB) correlations, and

- include the fuel assembly design enhancements.

3.4. NON-LOCA

Section 5 of the application (Reference 37) presents the revised non-LOCA accident analyses. The following changes are addressed:

- revision of Cycle 1 loading pattern
- implementation of WABA
- implementation of enhanced GRCA
- implementation of the RPG
- increase in $F_{\Delta H}$ limit
- increase to mid-grid and IFM height
- addition of RCCA drop time input
- incorporation of design changes to the AP1000 Revision 19 DCD as part of the finalization

All of the non-LOCA analyses of Chapter 15 have been revised for the Advanced First Core design changes.

Section 15 in Appendix B of the application (Reference 37) addresses proposed changes and updates to the AP1000 Revision 19 DCD accident analyses. The non-LOCA portion of the accident analyses is presented in Section 15, excluding Section 15.6.5. The general change items are described in the "Change Road Map" that precedes Section 15.0 in Appendix B of the application (Reference 37). The editorial changes include further clarification on how a loss of alternating current (ac) power is addressed for each event and the application of the protection system delay between reactor trip and turbine trip.

The system transient analyses are performed using the LOFTRAN computer code. The original and previous system analyses used a modified version of LOFTRAN referred to as LOFTRAN-AP (see Reference 9). LOFTRAN was modified to include the passive safety system of the AP1000 design to create LOFTRAN-AP. The passive plant changes have been combined with the latest version of LOFTRAN to create a single version of the code for use in AP1000 analyses and operating plant analyses. This latest version of LOFTRAN has been used for the Advanced First Core analyses.

The applicant added Section 15.0.11.6 to discuss the advanced nodal code (ANC) that is used in the analysis of the rod ejection event. The nuclear design parameters affected include the minimum SDM and the enthalpy rise hot channel factor ($F_{\Delta H}$).

3.5. LOCA

Section 15.6.5 in Appendix B of Reference 37 presents the changes and updates to the limiting small break loss-of-coolant accident (SBLOCA) and the limiting large break loss-of-coolant accident (LBLOCA) analysis described in Revision 19 of the AP1000 DCD. Other supporting documentation includes Reference 2 and Reference 10.

In Reference 37, the subsections describing SBLOCA and LBLOCA analyses remain largely unchanged outside of updates, including minor corrections, additional descriptions, minor update of analysis parameters, and a minor update of analysis results. The analysis conclusions were unaffected. All changes are described in the "Change Road Map," which precedes Section 15.6 in Appendix B of Reference 37. The only nuclear design parameter that was updated in the LOCA analyses was the enthalpy rise hot channel factor ($F_{\Delta H}$).

Reference 2 provides an additional discussion of the SBLOCA analyses, and Reference 10 discusses major changes to the LBLOCA analyses. These changes address concerns in Reference 11 about the potential impact on calculated peak cladding temperature (PCT) caused by fuel thermal conductivity degradation (TCD) with burnup in realistic emergency core cooling system (ECCS) evaluation models. Since all major changes are a result of accounting for fuel TCD, with the exception of the SBLOCA analysis update for a change in the limiting single failure, the technical evaluation is focused on the supplemental information contained in Reference 2 and Reference 10.

Comment [c2]: Editorial clarification to indicate that the SBLOCA analysis update was due to a change in the limiting single failure rather than TCD.

4.0 Technical Evaluation

4.1. Fuel Mechanical Design

The topical report outlines the physical fuel assembly design changes and evaluates the resultant changes to the applicable safety analyses. The applicant left out of the report analysis topics it identified as obviously unaffected. The staff compared the review topics identified in SRP Section 4.2 to the topical report and notes that the technical areas of fuel system damage and fuel rod failure are not directly analyzed.

The staff reviewed the changes to the AP1000 fuel assembly design identified in Reference 37. During the review of the documentation, the staff found that:

- The fuel assembly design changes (RPG and increased IFM/mid-grid heights) would not increase the stress, strain, or loading limits on the structural members compared to what has already been reviewed and approved by the staff previously.

- The contact points of the grids use the same design as the previously approved spacer grids and IFM ~~vane~~ grids.

Comment [c3]: Editorial correction – remove word "vane".

- No new materials are introduced for any of the design changes in the fuel assemblies.

- The design changes would not lead to increased fatigue cycles or changes in irradiation growth predictions.

Based on the above observations, the staff concludes that the fuel system damage analysis previously reviewed and approved in the AP1000 DCD Revision 19 remains applicable and no new analysis is necessary.

The staff reviewed the fuel assembly design changes following the guidance related to fuel rod failure found in SRP Section 4.2 and found that:

- The hydriding analysis previously approved would not be affected because the materials and processing have not been changed by the new design.

- Since the design and material for the fuel cladding and fuel pellet are not changed by the new fuel design presented in Reference 37, the respective analyses are not impacted (hydriding, collapse, overheating, and pellet/cladding interaction (PCI) and pellet/cladding mechanical interaction (PCMI)).

Based on the above observations, the staff concludes that the fuel rod failure analysis previously reviewed and approved in the AP1000 DCD Revision 19 remains applicable, and no new analysis is necessary.

4.1.1. GSI-191

The applicant provides an analysis of the protective grid design change on GSI-191 in Reference 37. The analysis relies on a comparison of the protective grid design as presented in the topical report compared to the GSI-191 testing performed in support of the AP1000 DCD Revision 19 and the PWR Owner's Group (PWROG) submittals. The testing programs covered a variety of operating parameters, break locations, and debris conditions. As a result of these test programs and a qualitative analysis of the changes to the fuel design, the applicant concludes that the majority of the pressure drop (ΔP) occurs across the bottom nozzle and protective grid of the fuel assembly.

In RAI CRR-013, the staff requested a description of all fuel design changes beyond the protective grid change presented in Section 2.6 of Reference 37. In response (Reference 5), Westinghouse described changes that included the mid grids and IFM grids previously approved in the AP1000 DCD Revision 19. These changes consisted of increasing the height of the grid straps to ~~increase assembly stiffness~~improve the grid to rod fretting margin and to improve the seismic performance (mid grid only). The overall design of the grids, including the

Comment [c4]: The main reasons for the grid changes were to improve the grid to rod fretting margin and to improve the seismic performance (mid grid only). The fuel assembly stiffness improvement was a derived benefit.

springs, remains the same. In Section 2.6 of Reference 37, the applicant states that the majority of the debris collects at the bottom nozzle and protective grid of the assembly and, therefore, the other components such as mid-grids and IFMs have a negligible impact on the GSI-191 testing results.

During an audit (Reference 14) of the Westinghouse GSI-191 testing program results for both the AP1000 (Reference 12) and PWROG (Reference 8), the staff reviewed photos of the resultant debris distribution, plots of the ΔP measurements during the test, and the test assembly configurations. The staff observed that the debris accumulated mostly on the bottom nozzle and the protective grid, with relatively minimal debris accumulating on the mid-grids and IFMs. The ΔP measurements recorded for the upper versus lower portions of the assembly confirmed this finding. These results are consistent with the statements provided in Reference 37.

Based on photos from previous test results provided during an audit, the staff notes that the debris depositing on the mid grids and IFM grids deposits on the leading edge, with no appreciable accumulation evident inside the grid itself. The response to RAI CRR-013 (Reference 5) also includes discussion regarding the pressure drop across the grid region compared with the overall assembly pressure drop, which demonstrates a relative lack of importance. Based on pressure drop comparison and the debris accumulation results, the staff agrees that the change in design to increase the grid heights of the ~~eMidgrids~~ mid grids and ~~eIFMs~~ IFMs is acceptable and does not require additional testing.

Comment [c5]: Change to mid grids and IFMs to be consistent with other places in the CRR.

In RAI CRR-014, the staff requested a comparison of the fuel design used in Reference 37 with the fuel designs used in the AP1000 DCD Revision 19, the AP1000 DCD Revision 19 GSI-191 testing (Reference 12), and the PWROG testing (Reference 8). In response (Reference 5), Westinghouse provided the dimensions for the components that make up the fuel assembly design for comparison. The response to RAI CRR-014, Table 1, shows that most of the fuel components have been used to support testing for either the AP1000 DCD Revision 19 fuel design or PWROG. The only components with a new design (as compared with operating plant fuel assembly designs or the previous AP1000 fuel assembly design) are the ~~eMid-Grids~~mid grids and ~~eIFMs~~ IFMs discussed previously. The components that lead to the largest pressure drop in the presence of debris-laden coolant are the bottom nozzle, protective grid, and bottom grid. These components are identical in the fuel design described by Reference 37 and the fuel design tested as part of the PWROG testing (Reference 8). Figure 1 of RAI response CRR-013 (Reference 5) shows an example pressure drop measured during the PWROG cold leg break testing. The results demonstrate the relative importance between the pressure drop measured across the bottom nozzle and RPG versus the total measured assembly pressure drop. Based on the information provided, the staff concludes that the testing performed on a similar fuel design as part of the PWROG testing program (Reference 8) contains identical key components, including the bottom nozzle, RPG, and bottom grid. These components are the locations at which debris beds form during GSI-191 testing leading to the largest contribution of the total pressure drop. The testing demonstrates that sufficient coolant would be available for long-term cooling, even in the presence of debris. Therefore, the staff concludes that the fuel

Comment [c6]: Change to mid grids and IFMs to be consistent with other places in the CRR.

design presented in Reference 37 would not lead to failures during long-term cooling as defined by GSI-191.

4.1.2. Fuel Seismic Response

In the staff's acceptance letter (Reference 16) on the review of Reference 1, the staff noted that additional information would be needed about the fuel seismic response at end of life (EOL) conditions because of reduced assembly stiffness caused by grid spring relaxation during irradiation. During the review process, the staff sent three RAIs related to fuel seismic response (CRR-024, CRR-025, and CRR-027). The responses to these RAIs were included in a supplemental information package submitted on April 30, 2013 (Reference 3).

In the supplemental information package for EOL seismic and LOCA calculations, the applicant determined EOL grid strength by relaxing the grid springs on sample grids and then performing impact tests following the same methodology used for beginning of life (BOL) AP1000 grid impact tests. The amount of relaxation chosen for the EOL condition tests was based on post irradiation experimentation (PIE) data from assemblies with burnups comparable to the AP1000 EOL burnups and with same-sized pins and RFA style grids. The applicant chose a gap size that exceeded the 95-percent confidence level of the mean based on these PIE data. This methodology for determining the amount of spring relaxation for EOL testing generally follows the guidance provided in SRP Section 4.2 Appendix A for BOL grids allowable crush load (P_{crit}). By basing the gapped cell methodology for EOL condition grids on a related approved methodology, the staff finds the approach acceptable.

Using both BOL grids and EOL surrogate grids, Westinghouse performed a total of 60 tests to calculate P_{crit}. Westinghouse identified two different collision mechanisms based on core geometry. These were identified as grid-to-grid and grid-to-~~barrel~~ core shroud collisions. Westinghouse modeled these collision mechanisms in testing by varying the back plate that the grids rested against in the test rig. For grid-to-grid collisions, a flat back plate was used, and for grid-to-~~barrel~~ core shroud collisions, a smaller back plate with one rounded edge was used. This modified back plate represented core locations on the periphery where the edge of a core ~~barrel~~ shroud projected partway past an assembly. During an audit on April 16, 2013 (Reference 15), the staff reviewed core drawings, the back plates used, and the testing setup. Based on the information provided, and confirmed during the audit of supporting materials, the staff finds that the methodology for determining P_{crit} follows the guidance provided in SRP Section 4.2 Appendix A and is acceptable.

Comment [c7]: Editorial correction. The AP1000 design uses core shroud, not core barrel.

Comment [c8]: Editorial correction. The AP1000 design uses core shroud, not core barrel.

Comment [c9]: Editorial correction. The AP1000 design uses core shroud, not core barrel.

In Section 3.0 of Reference 3, Westinghouse states that the modal frequencies for EOL conditions are lower than for BOL conditions. These lower frequencies result in higher grid impact loads. Westinghouse performed vibration tests using the same methodology as previously used in BOL tests, with the exception of using gapped grid cells for the EOL surrogate assembly. During an audit on April 16, 2013 (Reference 15), the staff saw the assembly used to generate the fuel assembly modal frequencies listed in Table 2 of Reference 3 and confirmed that the assembly was based on the fuel design described by the topical report. The staff accepts the modal frequencies listed in Table 2 of Reference 3 as

representative of the AP1000 CRR fuel design based on the use of previously approved testing methodology, full-sized assembly based on the new fuel design, and appropriately gapped grid cells to simulate EOL grids for the EOL assembly testing.

Section 5.0 of Reference 3 discusses the methodology used to take credit for damping of the fuel assembly vibration response. The core dynamic analysis for BOL conditions conservatively assumes still water and a damping ratio of 20 percent. Figure 4 of Reference 3 presents the data used to support the damping ratio. This follows the methodology used in previously approved Westinghouse submittals and the staff finds it is acceptable for the new fuel design in Reference 37.

The core dynamic analysis for EOL conditions takes credit for vibrational damping caused by the flowing water throughout the fuel assembly. Figure 5 of Reference 3 presents fuel assembly damping data in flowing and still water for various flow rates and temperatures. Figure 7 summarizes the statistical evaluation of the damping data, and Figure 8 presents the damping design curve. The damping curve includes adjustments for measurement uncertainty, temperature effects, and reserved margin.

Specific testing of the fuel design presented in the topical report did not generate the damping data presented in Reference 3. Instead, this data is based on other Westinghouse fuel design data and publicly available data for fuel designs from other vendors. The data presented in Table 3 demonstrates that the amount of damping is not strongly design dependent. The staff agrees that this demonstrates that flow induced vibration damping is not strongly dependent upon the grids, which are the primary concern for fuel failures during seismic events.

The damping curve presented in Figure 8 provides a damping coefficient as a function of flow velocity based on the data and uncertainties, as previously discussed. By crediting flow-induced damping in calculating the grid impact forces for EOL conditions, it is necessary to account for conditions in which the flow rate (and also the flow-induced damping credit) could be reduced. Westinghouse identified a pump coastdown event resulting from a loss of offsite power during a seismic event as leading to the most limiting conditions from the standpoint of flow-induced vibration damping. The methodology and analysis for the pump coastdown event is listed in Section 5.0 of Reference 3. By calculating the lowest pump flow rate occurring between the event initiation and RCCA insertion to the dashpot region, and using this flow rate as the basis for the damping coefficients used in the EOL grid analysis, the staff finds that the event is conservatively modeled and follows the guidelines found in SRP Section 4.2 Appendix A. Therefore, the staff finds this acceptable.

To confirm that the grids are the most limiting fuel assembly component, Westinghouse additionally analyzed the fuel rods to ensure that fuel rod fragmentation does not occur. Section 6.0 of Reference 3 lists the results, which demonstrate significant margin ~~to fuel rod~~ to fuel rod fragmentation.

Comment [c10]: Editorial revision.

In summary, the staff finds that the applicant follows regulatory guidance, where applicable, provides conservative methodologies when regulatory guidance is not available, and demonstrates that the AP1000 grid crush strength is not exceeded for BOL or EOL conditions.

Comment [c11]: Editorial revision.

4.2. Nuclear Design

Reference 37, Appendix A, Section 4.3, "Nuclear Design," presents the changes to the design bases for the AP1000 nuclear design. The nuclear design must ensure that the SAFDLs will not be exceeded during normal operation, including anticipated operational occurrences (AOOs), and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary (RCPB) or impair the capability to cool the core.

4.2.1. Reactivity Control and Shutdown Margin

A significant change highlighted in Reference 37 is in the GRCA design. This is important to the nuclear design because of the change in the absorber material from Ag-in-Cd to tungsten. A reference to WCAP-16943-P-A (Reference 7) was added to support the use of the enhanced GRCA rodlet design as part of the AFC nuclear design. The NRC approved the use of this GRCA design for the AP1000, as stated in the conclusions of the staff's SER. The tungsten GRCAs are not safety-related components and do not adversely affect SAFDLs, as discussed in the staff's SER.

Comment [c12]: Approved version of Topical Report should be referenced for consistency with citation of previously approved Topical Reports.

The staff also observed in Table 4.3-1 of Reference 37 a first core design change from Pyrex burnable absorber to WABA. The NRC previously approved WABA for use in Revision 19 of the DCD and also for general use in PWR cores in the staff's SER for Reference 17. Therefore, the staff finds this to be acceptable.

In Section 4.3.2.4, "Control Requirements," (Reference 37 Appendix A) two paragraphs were added explaining that the GRCAs are not credited in the calculation of available shutdown margin, despite being released into the core upon reactor trip. The credit of only the RCCAs allows the continued use of the 7-percent uncertainty allowance for the credited rod worth. It is further stated that the GRCAs may be credited after various conditions are met after the reactor is shut down. Since the GRCAs are not relied upon to safely shut down the reactor and, therefore, not credited in the available shutdown margin calculation, the staff finds this to be conservative and acceptable. Since SDM requirements remain to be met, GDC 27 is satisfied. In Section 4.3.2.4.5, "Rod Insertion Allowance," (Reference 37, Appendix A), the paragraph was revised to emphasize how the pre-trip control rod insertion can affect the available trip rod worth and how the GRCA positions also can have a small effect on the worth of the RCCAs. It explains that the most limiting allowed control rod insertion is implicitly included in the calculated trip rod worth and total power defect values reported in Table 4.3-3 (Reference 37, Appendix A).

In reviewing the changes to Table 4.3-3 in Reference 37, Appendix A, the staff observed that, despite there being less of a power defect to overcome, the rods are worth less. This translates into less SDM available, as seen in Table 4.3-3. However, there is still significant margin to the requirement. Therefore, the staff finds this acceptable, as GDC 27 continues to be met. It is

also noted again that the GRCAs are not credited in the SDM calculation, which is a conservative assumption.

Section 4.3.2.4.16, "Load Follow Control and Xenon Control," (Reference 37, Appendix A) was modified to elaborate on the use of soluble boron above 30 percent rated thermal power (RTP) to accommodate large load changes. It is implied that the automatic mechanical shim (MSHIM) typically operates without the need for boron control, but in this case, it is explained that soluble boron may optionally be used during MSHIM operation to maintain optimum GRCA positioning and to minimize use of the more reactive RCCAs. The staff finds the allowance of soluble boron use during load follow control maneuvers acceptable since this is a change that was implemented to allow more operational flexibility.

4.2.2. Reactivity Coefficients

The staff reviewed changes to the reactivity coefficients as presented in Table 4.3-2 of Reference 37. Table 4.3-2 shows only slight changes to the typical best estimate reactivity coefficients (i.e., they are more negative, which is conservative). The design limits for the reactivity coefficients remain unchanged.

As part of an audit (Reference 14), the staff reviewed various rod ejection analyses that form the conclusions drawn in DCD Section 15.4.8 (see Section 4.~~5~~6.26 of this report) discussing rod ejection accident analyses in more detail. In all of the analyses reviewed, a multiplier was used to adjust the effect of the Doppler feedback, which directly impacts the transient response predicted by ANC, the three-dimensional (3-D) spatial kinetics code used in all of the rod ejection simulations. Westinghouse explained that this multiplier operates by conservatively reducing the change in the fast fuel absorption cross section due to changes in the effective fuel temperature, which in turn conservatively increases the magnitude of the power spike seen after the simulated rod ejection. ~~Before the submittal of Reference 1, the assumed multiplier was [ ]a,c. Westinghouse explains that the multiplier was increased to account for the unforeseeable impact caused by incorporation of fuel TCD effects into the fuel temperature models that are a part of ANC.~~ Westinghouse ~~further~~ explained that it builds a minimum [ ]a,c percent reserve margin into the Doppler feedback for ~~all of its safety~~ the rod ejection transient consistent with the approved methodology in Reference 27~~calculations to accommodate for cycle-to-cycle variations—an administrative requirement~~. This can be compared to the Doppler feedback coefficient by reducing the change in the fast fuel absorption cross section due to the change in the effective fuel temperature by [ ]a,c percent or by multiplying by [ ]a,c. ~~Moving from [ ]a,c to the chosen~~The analysis documented in Reference 37 used a value of [ ]a,c ~~still satisfies this administrative limit, but it also allows Westinghouse to gain back margin. Although the change in the multiplier is nonconservative with respect to the previous value, it is still an overall~~which is a conservative modification to the Doppler coefficient relative to the licensed methodology and 10 CFR Part 50 GDC 11, "Reactor Inherent Protection," remains satisfied.

Comment [c13]: Typographical error corrected to cite correct section. There was a break in the numbering sequence in sub-section 4.6 starting with 4.6.13

Comment [c14]: Added text for consistency with description of adjustment to Doppler feedback given in WCAP-15806-P-A (Ref. 27)

Comment [c15]: Deleted sentence beginning with "Before the submittal of Reference 1" since this discusses scoping analyses that are not relevant to the actual analysis performed to support the CRR.

Clarified the statement "builds a minimum [ ]a,c reserve margin into all of its safety calculations" to indicate that [ ]a,c margin is applied to the Doppler feedback per the approved methodology in Ref. 27. Further clarification to describe adjustment to Doppler feedback made by the change to fast fuel absorption cross-section consistent with text in Ref. 27.

Also made Doppler multipliers / percent changes bracketed as proprietary since they are marked proprietary in WCAP-15806-P-A.

4.2.3. Peaking Factors

Table 4.3-2 of Reference 37, Appendix A, shows no change to the heat flux hot channel factor

(F_Q); however, the staff observed an increase to all of the enthalpy rise hot channel factors ($F_{\Delta H}$)

in the table. Note (g) states that the hot channel factors presented in Table 4.3-2 are for fully inserted rods, which is a conservative assumption since this maximizes peaking.

DCD Table 4.4-1 (Sheet 1 of 2) was updated to reflect the reduction of the two minimum departure from nucleate boiling ratio (DNBR) values calculated at nominal conditions; these values remain well above the minimum DNBR limit prescribed for design transients.
As stated in Section 3.5 and Section 5.2 of Reference 37, the revised Chapter 15 transient analyses were performed assuming the increased nuclear design $F_{\Delta H}$ limit, but this is with respect to the previous nuclear design value and not the value assumed in the Chapter 15 transient analysis for DCD Revision 19, which included additional margin. A review of the revised Chapter 15 subsections given in Appendix B of Reference 37 found that an "AP1000 Core Reference Report DCD (Revision 19) Change Road Map" before each subsection summarizes all of the changes for the respective subsections. The report notes that the nuclear design $F_{\Delta H}$ limit increase is documented as a change for each transient scenario in which $F_{\Delta H}$ is relevant. The revised Table 15.6.5-4, "Major Plant Parameter Assumptions Used in the Best-Estimate Large-Break LOCA Analysis," documents further verification of the nuclear design $F_{\Delta H}$ limit increase in the Chapter 15 transients. However, it is noted that the previously assumed $F_{\Delta H}$ limit for the LBLOCA analysis was actually higher in the DCD Revision 19 analysis; the update to the limit is now consistent with the design value based on the revised nuclear design and is appropriate for use in the Chapter 15 transient analyses.

4.2.4. Vessel Fluence

Reference 37, Appendix A, Section 4.3.2.8, "Vessel Irradiation," describes the methods and analyses used to predict reactor pressure vessel (RPV) irradiation, and refers to Section 5.3 of the DCD for a more detailed discussion that includes the irradiation surveillance program. The staff identified a potential impact to the projected EOL RPV fluence calculation with the change to the AFC and subsequent core loading patterns. During an audit (Reference 14), the various AFC relative assembly powers in Reference 37 (both for the first cycle and for the equilibrium cycle) were compared to the original DCD relative assembly powers for the equilibrium core. The DCD relative assembly powers for the equilibrium core are higher than the AFC low leakage pattern (L3P) design on the periphery, which indicates that the vessel fluence will be lower for the AFC loading pattern since these peripheral assemblies carry the most weight with respect to fluence accumulation within the reactor pressure vessel. This follows with the concept of the L3P, which is used to enhance neutron economy and minimize neutron leakage from the core. Furthermore, the L3P is designed to look like the equilibrium cycle. Therefore, it is expected that the equilibrium cycle for the updated core design loading will produce similar relative assembly powers. Consequently, Revision 19 of the DCD contains a more conservative estimate for vessel fluence compared to the Reference 37 changes. Therefore, the design basis nil-ductility transition temperature that factors into the RT_{NDT} limit in Table 5.3-3 of the DCD remains conservative, along with the pressure and temperature limit curves and the low temperature overpressure protection system setpoints referenced in the Chapter 16 technical specifications. Furthermore, the requirements of 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," remain satisfied.

4.2.5. Spent Fuel Pool Criticality

During the Phase 1 audit, the NRC held discussions with Westinghouse related to the impact of the AFC and subsequent core reload designs on the SFP criticality safety analysis performed in Section 9.1.1 of the DCD. The staff's main concern was with the fuel assembly limiting depletion characteristics, including the impact on the limiting axial burnup profile assumption. To capture the information discussed, the staff issued RAI CRR-001 asking Westinghouse to discuss how the change to the AFC and subsequent core designs—in addition to how the designs are managed from cycle to cycle—impact the assumptions and conclusions made in Reference 18. In the response to RAI CRR-001 (Reference 6), Westinghouse indicated that it took into account the AFC fuel design characteristics, burnable absorber loading and control rod management schemes, and corresponding axial burnup profile shapes in the current analysis. As a result, the conclusions of the SFP criticality safety analysis are unchanged, and the criteria in 10 CFR 50.68, "Criticality accident requirements," remain satisfied. Westinghouse also noted that a minor discrepancy was identified and entered into the Westinghouse Corrective Action Process. The discrepancy was with the diameter of the GRCA absorber assumed in the criticality analysis, which is inconsistent with the final design of the GRCAs used in the newer AFC design. The assumed diameter is smaller than the actual absorber diameter, which would have a nonconservative impact on the criticality analysis. In the revised analysis performed by Westinghouse, as indicated by the response to RAI CRR-001 (Reference 6), it is demonstrated that the impact of the revised GRCA rodlet diameter on the spent fuel pool criticality analysis is negligible (Reference 28). The staff agrees that the design changes specified in Reference 37 continue to meet the regulatory requirements regarding spent fuel pool criticality by following the guidance provided in SRP 9.1.1.

The staff also observed that the reference to APP-GW-GLR-029P, Revision 3, "AP1000 SFP Storage Racks Criticality Analysis," was deleted and replaced by a reference to the CASMO-4 user's manual (new Reference 53 in Section 4.3). It is explained that this was done because only the criticality methods are described in Section 4.3.2.6.1 and not the analysis. Section 9.1.6 describes the analysis results. Since this reference change did not introduce any new methods and is just for clarification, the staff finds this to be acceptable.

4.2.6. Technical Specifications

Section 4.3.2.2.6, "Limiting Power Distributions," of Reference 37, Appendix A, mentions relaxed axial offset control (RAOC), which defines the envelope for allowable axial offset during reactor operation. Westinghouse identified conforming changes to LCO 3.2.1 and LCO 3.2.3 in the TS and corresponding TS bases that result from a change from RAOC to constant axial offset control (CAOC). CAOC is operationally more restrictive than RAOC and, consequently, results in added conservatism or margin with respect to minimizing xenon oscillations and subsequent effects on power distribution. Therefore, the staff finds the use of CAOC instead of RAOC to be acceptable and 10 CFR Part 50, GDC 12, "Suppression of Reactor Power Oscillations," remains satisfied. Reference 5 in the TS bases for LCO 3.2.1, which was previously reviewed and approved by the NRC, discusses the differences between RAOC and CAOC.

4.2.7. Codes and Methods

The CRR includes various updates to the referenced codes and methodologies used in the nuclear design. Reference 19 was added to allow use of PARAGON instead of PHOENIX-P to generate all of the cross sections used in the nuclear design. In Section 4.0, entitled, "Conditions and Limitations," of the staff's SER for PARAGON, Item 1 states that "the PARAGON code can be used as a replacement for the PHOENIX-P lattice code, wherever the PHOENIX-P code is used in NRC-approved methodologies." Since the staff previously approved PHOENIX-P for use in nuclear design analyses, the staff finds the use of PARAGON as an alternative to PHOENIX-P to be acceptable per the PARAGON SER conclusions (Reference 20).

The addition of Reference 71 provides additional information describing the cross section data used in the PHOENIX-P code. Since this is supplemental information, the staff finds the addition of the reference is acceptable.

A reference for WCAP-16045-P-A, "Qualification of the NEXUS Nuclear Data Methodology" (Reference 22), was added to allow use of the NEXUS nuclear data methodology, which passes nuclear data to the 3-D ANC. The staff approved this methodology for performing calculations on uranium-fueled PWRs, as documented in the corresponding SER conclusions (Reference 21). Consequently, the staff finds the use of this code is acceptable for applicable nuclear design analyses for the AP1000 design.

WCAP-10965-P-A, Addendum 2-A, "Qualification of the New Pin Power Recovery Methodology" (Reference 23), was added to the references to reflect an updated methodology to be used along with the ANC. The staff's SER states that this methodology is acceptable as long as the nuclear data generated as input to ANC originates from the PARAGON and NEXUS code systems. During an audit (Reference 14), the staff asked if there were any instances in which PHOENIX-P is used instead of PARAGON to generate data for ANC since the SER conclusions for the pin power recovery methodology only give approval for use of the methodology with the PARAGON/NEXUS/ANC system. Westinghouse stated that 1-D parameters, such as leakage factors, are input into the ANC directly from PHOENIX-P. It explained that instead of re-generating these 1-D data using PARAGON, the data from PHOENIX-P was used to avoid additional calculations. Westinghouse stated that it plans to move solely to PARAGON for the nuclear design at a later date. It further explained that the transport methods used to generate this data are different between PHOENIX-P and PARAGON (i.e., discrete ordinates vs. collision probability), so the data generated will be different, but are expected to be consistent. With the approval of the pin power recovery methodology described in Reference 23, the limitations and conditions in the NRC SER apply only to the use of ANC for rodded and unrodded pin power predictions. RAI CRR-002 was issued to confirm that PHOENIX-P is being used appropriately, and in the response (Reference 5), Westinghouse stated that the use of PHOENIX-P data as input to ANC is only used for nonfuel radial reflector regions located outside of the active fuel in which no pin power recovery calculations are performed. Therefore, the staff finds the limited

use of PHOENIX-P for the AP1000, as described above, continues to meet the regulations identified in Section 4.3 of SRP/NUREG-0800 and is therefore acceptable.

4.3. Thermal Hydraulic Design

In the CRR, Appendix A, Section 4.4, "Thermal and Hydraulic Design," the applicant proposed changes to the AP1000 thermal hydraulic design as compared with Revision 19 of the DCD. The principal function of the thermal hydraulic design is to provide adequate heat removal capability to ensure that the SAFDLs will not be exceeded during normal operation and transients. The applicant proposed the following five changes related to the thermal hydraulic design:

(1) replace the DNBR limits of 1.22 and 1.21 with a single conservative limit of 1.25

(2) add the ABB-NV and WLOP (ABB-NV modified for low pressure) correlations for certain conditions

(3) adjust F factors used for the WRB-2M, ABB-NV, and WLOP CHF correlations,

(4) change the enthalpy rise hot channel factor ($F_{\Delta H}$) limit from 1.65 to 1.72 and the corresponding factor for the Revised Thermal Design Procedure (RTDP) from 1.590 to 1.654

(5) revise the description to clarify the minimum incore nuclear detector instrumentation system requirements.

In addition, the thermal conductivity of the uranium dioxide fuel described in Section 4.4.2.11.1 is dependent on the fuel temperature without considering the degradation of thermal conductivity from the burnup effect. The staff included a review of the uranium dioxide (UO_2) fuel thermal conductivity degradation TCD impact on heat transfer properties with increasing fuel exposure.

In Section 4.4.1.1.2, the RTDP design limit DNBR values of 1.25 for the typical cell and 1.25 for the thimble cell replace the DNBR limits of 1.22 and 1.21, respectively. The DNBR limits were for those transients that use the VIPRE-01 computer program in Subsection 4.4.4.5.2 and the WRB-2M correlation in Subsection 4.4.2.2. These values are also changed in Table 4.4-1 and may be revised when plant-specific uncertainties are available. The staff finds this change to the higher DNBR limit of 1.25 to be conservative and, therefore, acceptable.

The thermal hydraulic CHF calculation of the AP1000 fuel uses the WRB-2M CHF correlation. The WRB-2M correlation applies to the Robust Fuel Assemblies planned to be used in the AP1000 core. The applicable range of parameters for the WRB-2M correlation is described in Section 4.4.2.2.1 of the CRR. Whenever the condition exists in which the WRB-2M is not applicable, other applicable CHF correlations are used. In the heated region below the first

mixing vane grid, the W-3 CHF correlation, which does not take credit for mixing vane grids, was used to calculate DBNR values. In addition, the W-3 correlation was applied in the analysis of accident conditions when the system pressure was below the range of the WRB-2M correlation.

In Section 4.4.2.2.1, the applicant added two new CHF correlations (ABB-NV and WLOP) that augmented the W-3 correlation as a configuration enhancement to the DNBR calculation. Both ABB-NV and WLOP CHF correlations were developed based on rod bundle CHF test data and provide more accurate prediction of the DNBR margin than the W-3 correlation, which was developed with tube data. The uses of these two correlations have been approved by the NRC, as described in addendums to WCAP-14565-P-A. Addendum 1-A, "Addendum 1 to WCAP-14565-P-A Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code" (Reference 32), addresses the ABB-NV CHF correlation, and Addendum 2, "Addendum 2 to WCAP-14565-P-A Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications" (Reference 33) addresses the extended application of CHF ABB-NV correlations and the WLOP CHF correlation for PWR low pressure applications. For instance, the ABB-NV correlation is used when conditions, in the fuel region below the first mixing vane grid, cause the DNB to become limiting with a severely bottom-skewed axial power distribution, such as rod withdrawal from a subcritical accident; whereas, the WLOP correlation was added for the analysis of Hot Zero Power Steamline Break event in which low flow and low pressure conditions exist. The replacement of the W-3 correlation with the ABB-NV and WLOP correlations are for consistency with the AP1000 fuel design enhancements and improvement in the DNBR calculation. Since the use of the ABB-NV and WLOP correlations have been approved by the NRC, the staff concludes that the changes are acceptable.

Section 4.4.2.2.2 describes the definition of DNBR as the predicted critical or DNB heat flux divided by the actual local heat flux. Since the W-3 CHF correlation is augmented by the ABB-NV and WLOP correlations, the description related to the W-3 correlation is deleted, which the staff finds acceptable. The predicted DNB heat flux is calculated by the equivalent uniform DNB heat flux predicted by a DNB correlation divided by the flux shape F factor to account for the nonuniform axial heat flux distribution. Section 4.4.2.2.2 states that adjusted F factors are used for the WRB-2M, ABB-NV, and WLOP correlations. An adjustment factor is applied to the F-factor for the calculation of the DNB heat flux with nonuniform axial power shape. The staff had evaluated the adjustment factor used for the WRB-2M correlation. As stated in a letter of February 3, 2006 (Reference 34), the staff concluded that the effect of the adjustment factor is to lower the DNBR, which provides more conservatism in the plant reload evaluations than the current correlation. The DNB margin retained in the safety analysis DNBR limit remains unaffected. Therefore, the staff finds the use of the adjustment factor is acceptable.

The WRB-2M correlation, documented in WCAP-15025-P-A, was developed for the modified low pressure drop (MLPD) mid-grid design and the modified intermediate flow mixer (MIFM) grid design. The AP1000 mid-grid and IFM grid designs have been enhanced relative to the MLPD and MIFM grid designs by increasing the grid strap heights (by about 50 percent) to improve fretting wear resistance of the AP1000 fuel assembly. Since the grid enhancement changes are

limited to the increase of the grid strap heights, the applicant contends that the DNB correlations remain applicable to the AP1000 fuel with enhanced grid design. In Section 4.5 of the AP1000 CRR, the applicant states that the AP1000 mid and IFM grids have the same DNB performance as the MLPD and MIFM grids because of the following:

- [ ]a,c (MLPD grid height is 2.25 inches (5.7 centimeters) and MIFM height is 0.66 inches (1.7 centimeters)), [ ]a,c.
- The AP1000 grids maintain the same mixing vane design and mixing vane pattern as the MLPD and MIFM grids, which is the dominant DNB parameter.
- The important DNB performance parameters in the Westinghouse Fuel Criteria Evaluation Process focus only on lateral grid parameters (not axial parameters like grid height).

The applicant therefore concludes that the DNB correlations are applicable to the AP1000 fuel having the enhanced mid grid and IFM grid design. In the staff audit of WCAP-17524-P (Reference ~~36~~37), the applicant provided a figure of the measured-to-predicted CHF ratios versus grid strap heights to demonstrate that grid strap height does not have an effect on DNB performance. The figure was based on the DNB tests for the WNG-1 DNB correlation, which was described in topical report WCAP-16766-P-A (Reference ~~37~~42) and has been approved by the NRC. The DNB tests were performed with test assemblies with various grid strap heights but same mixing vane design. The figure shows no trend of the measured-to-predicted CHF ratios against the grid strap height, indicating that grid strap height has no significant effect on the DNB performance. The staff concludes that the increased strap height used in the fuel design presented in Reference 37 has insignificant effect on the DNB performance, as supported by a previously reviewed and approved topical report and, therefore, the WRB-2M correlation is applicable.

Comment [c16]: For consistency with numbered list in Section 7.0 Reference

Comment [c17]: WCAP-16766-P-A is not listed in Section 7.0 Reference. It should be added to Section 7 as Reference 42.

Section 4.4.2.11.2 describes a formula for calculating the radial power distribution in the UO_2 fuel rods with a radial power depression factor F, which is determined using the radial power distributions predicted by the neutron transport theory code LASER. A second equation was added to account for annular fuel pellets that required a mathematical correlation applied to the first equation. The basis for adding this equation is that the AP1000 fuel rod design also may include annular fuel pellets in the top and bottom 20 centimeters (8 inches) of the fuel stack, as described in Subsection 4.2.2.1., "Fuel Rods," of the final safety analysis report. The staff finds the proposed change acceptable because the annular cylindrical relationship incorporated in the general heat conduction equation was confirmed by mathematical derivation.

In Sections 4.4.4.3.1 and 4.4.4.3.2, the applicant revised the "enthalpy rise hot channel factor" limit, at RTP condition, from 1.65 to 1.72, as required for the AFC configuration. As a result, the corresponding full-power enthalpy rise hot channel factor for the RTDP increased from 1.590 to 1.654, which included a statistical measurement uncertainty of 4 percent (1.72/1.04 = 1.654).

Therefore, the revised AP1000 thermal hydraulic and safety analyses were performed assuming an "enthalpy rise hot channel factor" limit of 1.654. As a result of increasing the enthalpy rise hot channel factor, the hot channel void fractions listed in Table 4.4-2 also are increased. The staff finds that the changes are necessary for the AP1000 AFC configuration and are acceptable. However, the staff notes that analyses must continue to follow the approved design methodologies using the revised enthalpy rise hot channel factor, as required, to change TS parameters defined in the COL report.

In Section 4.4.6.1, the applicant discusses the incore instrumentation system detector operability requirement permitted to perform the primary function of providing a 3-D flux map of the reactor core. The 3-D flux data is used to calibrate the neutron detectors used by the protection and safety monitoring system, as well as to optimize core performance. To clarify the minimum number of detector segments required in each core quadrant, the applicant described the incore instrumentation thimble assembly as consisting of multiple fixed incore detector elements that start at the top of the active fuel and are sequentially placed along the vertical axis to the bottom of the active fuel of the fuel assembly. In addition, a detector segment was defined as the difference in signal from two operable detectors within the same assembly. With a total of 42 incore instrument thimble assemblies placed throughout the core, the calculation algorithms measures the signal from each operable detector segment. The minimum number of incore assemblies detectors required for operating the system is at least 75 percent operating detector segments during the initial power distribution measurement required in each operating cycle, and at least 40 percent operating detector segments following the cycle initial power distribution measurement. A minimum of 15 operating detector segments in each quadrant with at least six detector segments below the core mid-plane and six detector segments above the core midplane in each quadrant is required both before and following the cycle initial power distribution measurement. The basis for the change is to eliminate misinterpretation of the requirement. For instance, a configuration could exist in which each quadrant has 30 operable detectors but less than 15 operable detector segments, as needed, based on the analysis. Since this change of the incore instrumentation description provides clarification to prevent misinterpretation of the operability requirement, the staff finds it acceptable.

4.4. FUEL THERMAL CONDUCTIVITY DEGRADATION (TCD) EVALUATION OF DCD SECTION 4.4

AP1000 DCD Section 4.4.2.11, "Fuel and Cladding Temperatures," discusses the principal factors employed in the determination of the fuel pellet temperature distribution over the fuel rod lifetime, such as gap size, internal gas pressure, gas composition, pellet density, fuel clad surface condition, and UO_2 thermal conductivity. The fuel pellet thermal conductivity is by far the primary factor that influences the pellet temperature profile. Since the UO_2 fuel pellet thermal conductivity is a function of fuel burnup that leads to TCD, as described in NRC Information Notice 2009-23 (Reference 34), "Nuclear Fuel Thermal Conductivity Degradation," the staff submitted several RAIs related to the calculation method used in the performance analysis and design (PAD) 4.0 code and the impact on the fuel pellet temperature distribution and centerline temperature related to thermal hydraulic design. The applicant developed the

"PAD 4.0 TCD" code version as an assessment tool to evaluate the effects of fuel burnup TCD for the operating fleet and is also using it in the evaluation of the AP1000 fuel design presented in the CRR.

It should be noted that the staff is not performing a standalone safety evaluation of the PAD 4.0 TCD code (also known as the "PAD 4.0 TCD assessment tool"), but is evaluating its use for AP1000 and the impact on the fuel pellet heat transfer reduction and center line temperature rise prediction as a function of fuel burnup for the fuel design and core design presented in the CRR.

During the audit and evaluation of the CRR Section 4.4 proposed changes, the staff had concerns about the PAD 4.0 TCD code computation techniques of the TCD and its impact on the final AFC limit designs. The staff concerns focus on the: (1) TCD calculation with increasing bundle exposure, and (2) TCD effects on the thermal hydraulic and safety analysis. The staff submitted two RAIs—CRR-010 and CRR-011—each of which contain a series of questions pertaining to the TCD effects on the thermal hydraulic analysis.

In RAI CRR-010, the staff requested a discussion of the development and validation of the TCD model used in the interim version of PAD 4.0 TCD, including all coefficients of the TCD equation. The applicant provided the fuel thermal conductivity equation as modeled in the licensed PAD 4.0 with an explanation of the equation's components, including a fuel exposure dependent function, f(Bu), which is presently deactivated pending NRC approval of this function for PWR licensed PAD 4.0. Therefore, the licensed PAD 4.0 fuel performance models described in WCAP-15063-P-A, "Westinghouse Improved Fuel Performance Analysis and Design Model (PAD 4.0)," do not address the impact of fuel thermal conductivity degradation in the thermal hydraulic analysis.

To assess the TCD impact on the fuel, the burnup dependent function, f(Bu), defined by a set of non-zero fuel exposure dependent coefficients, was incorporated into the PAD 4.0 code. This model was based on a TCD with burnup model recommended by the Halden Project Report, "Thermal Performance of High Burnup Fuel—In-Pile Temperature Data and Analysis," (Reference 35), which was previously integrated into the Westinghouse STAV 7.2 fuel performance code described in WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors—Supplement 1" (Reference 36). The STAV 7.2 code is employed in European reactors and approved by the NRC in the United States for boiling-water reactor (BWR) applications. The applicant's initial effort in developing a tool for assessing TCD effects resulted in a small conservative adjustment to the TCD impact on fuel thermal conductivity in the STAV 7.3 code version that yielded coefficients used in the PAD 4.0 TCD code.

In the development of the PAD 4.0 TCD assessment tool, the applicant considered two models: (1) the STAV TCD model based on the Halden TCD model, and (2) the modified Nuclear Fuels Industries (NFI) model, which is currently integrated into the FRAPCON 3.4 NRC fuel performance code described in NUREG/CR-7022 Volume 1, "FRAPCON–3.4: A Computer Code for the Calculation of Steady-State Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup," and Volume 2, "FRAPCON–3.4: Integral Assessment," (References 29 and 30).

The applicant selected the STAV code model for the PAD 4.0 TCD because of the similarity in the modeling forms for fuel thermal conductivity in both the PAD 4.0 and STAV 7.2 codes. Although the STAV 7.2 code was developed for European BWR reactor designs, the fuel thermal conductivity property is not dependent on reactor design but is a function of fuel material, temperature, fuel density, burnup, and thermal expansion. Therefore, the STAV TCD model for f(Bu) is applicable to PAD 4.0 TCD PWR analysis.

The applicant validated the PAD 4.0 TCD assessment tool by comparing the PAD 4.0 TCD-predicted centerline temperatures with the measured fuel centerline temperatures from the Halden test reactor database, with a range of fuel burnups. The fuel centerline temperatures predicted by PAD 4.0 TCD were found to be in good agreement with measured values, with no further adjustment to the TCD model coefficients. The applicant provided four plots of thermal conductivity versus temperature at burnups of 0, 20, 40, and 65 gigawatt-days per metric ton of uranium (GWd/MTU) with or without the burnup coefficients default option enabled. As expected, the staff confirmed the significance of the fuel thermal conductivity as a function of fuel burnup and noted the following observations: (1) as fuel exposure increases, there is a significant downward shift in fuel thermal conductivity, and (2) the maximum shift occurs at low temperature and virtually converges above the temperature of 2,500 °C. Therefore, the staff concludes that the data confirms the fuel thermal conductivity is a function of fuel burnup that needs to be addressed in the safety analysis.

In response to a staff request, the applicant provided plots demonstrating the differences between the approved PAD 4.0 and the "PAD 4.0 TCD assessment tool" thermal conductivity models. For each plot, the difference between the measured and the predicted (M-P) centerline temperature (°F) was plotted against fuel burnup (MWD/MTU). The staff observed that the plot without TCD compensation showed a linear increase in the difference between the measured and predicted temperature with a burnup between 0 and approximately 75,000 MWD/MTU, with a maximum temperature difference of approximately 204 °C (400 °F). Above 75,000 MWD/MTU, the difference in temperature (M-P) decreases with increasing fuel burnup. Whereas, the plot using PAD 4.0 TCD assessment tool showed almost no change in the difference temperature over the full range of fuel burnup. Therefore, the PAD 4.0 TCD assessment tool is consistently accurate with the measured data at any fuel exposure. The staff concludes that the PAD 4.0 TCD assessment tool provides an accurate temperature profile across the full range of approved fuel exposure with a high degree of reliability.
The applicant provided a description of the PAD 4.0 fuel performance code thermal model calibration and validation process that included three initial NRC-sponsored Halden tests followed by additional experiments conducted to further understand fuel properties as a function of burnup and to reduce the uncertainties in the model.

Each of the three Halden tests (IFA-431, IFA-432, and IFA-513) consisted of six test rodlets fabricated at the PNNL facility. To accommodate the initial pellet-clad gaps used in commercial fuel rod design, the test rod fabrication parameters were designed with a range of initial pellet-clad gaps of 2.2 mils to 15.1 mils. In addition, the IFA-431 and IFA-432 test rods were pre-pressurized with one atmosphere helium. The IFA-513 test rods had a range of pre-pressurization conditions that included 100 percent helium at one or three atmospheres and

a mixture of helium and argon at one atmosphere. All test rods were instrumented with thermocouples at the top and bottom of the fuel stack.

The data obtained from the testing were used in PAD 4.0 thermal model calibration and development at low burnup conditions. This included online fuel centerline temperature measurements as a function of burnup, up to 5 GWd/MtU. Also, the data obtained, because of initial gap conditions simulated the range of gap conditions expected in Westinghouse PWR operation, were considered applicable to the full operating fuel burnup range. In addition, the relatively low burnup data provided an opportunity to minimize the uncertainties associated with cladding creep, fission gas release, and thermocouple decalibration.

To obtain data of the irradiation effects on thermocouple response behavior at extended burnup conditions, the IFA-432 test rod was irradiated in excess of 45 GWd/MtU and, from this data, the Halden project developed decalibration correction factors to compensate for the thermocouple response. The PAD 4.0 thermal model was further validated using the higher burnup data for the IFA-432 test rods. Although the measured versus predicted fuel temperature comparisons for the extended burnup data did not show a clear trend with burnup, the NRC approved the PAD 4.0 thermal model without the fuel thermal conductivity degradation function.

Subsequent to these NRC-sponsored Halden tests, additional experiments have been conducted to further understand fuel properties as a function of burnup. To avoid uncertainties associated with thermocouple decalibration, two alternative approaches were considered. First, tests were run with annular pellet-fueled rods with expansion wire thermometers in the annulus. This instrumentation is not subject to decalibration under irradiation, and these tests provided the first definitive results confirming TCD. To improve the expansion wire thermometer results by reducing the potential error caused by pellet fragment interference with the wire, Westinghouse performed additional tests using re-fabricated commercially irradiated fuel rods into test rods instrumented with new thermocouples. These tests provided high quality temperature measurements at high burnup but introduced an additional uncertainty in the fuel rod conditions because of re-fabrication. However, the composite data results of tests discussed above were used to validate the PAD 4.0 TCD model.

For these reasons, the staff released IN 2011-21, Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation" (Reference 11), which notified the industry of the information related to the impact of irradiation on fuel thermal conductivity and its potential to cause errors in realistic emergency core cooling system evaluation models.

The burnup-dependent TCD equation from STAV 7.3 is used in the interim version of PAD 4.0, called PAD 4.0 TCD. The STAV 7.3 code is approved for BWR fuel design analysis and is designed to handle fuel with gadolinia as a burnable absorber. The AP1000 PWR fuel design may include axial blankets (fuel pellets of a reduced enrichment), annular fuel pellets in the top and bottom 20 centimeters (8 inches) of the fuel stack (fully enriched or partially enriched), and burnable absorber (ZrB_2-coated fuel pellets or fuel pellets containing gadolinia oxide mixed with uranium oxide). In RAI CRR-011, the staff requested that the applicant address the impacts

that this fuel design may have on the thermal conductivity correlation, and discuss the applicability of the TCD equation from STAV 7.3 for the integral fuel burnable absorber (IFBA) coated pellets.

The applicant states that the "Halden test rods used for thermal model validation are applicable to the full range of fuel pellet types used in the AP1000 PWR fuel rod design," including the UO_2 material properties for uranium-235 enrichments and natural enriched uranium in the axial blanket pellets. The effects of enrichment on pellet radial power distributions are accounted for as input to the PAD code. The PAD 4.0 TCD model validation is, therefore, applicable to the range of pellet types used in the axial blanket region.

The IFBA pellets have a coating of ZrB_2 on the outside pellet surface. This coating has only a negligible thermal resistance and is conservatively modeled in PAD as a small increase in the fuel diameter. The coating has no impact on the temperature gradient across the fuel pellet, and the Halden test results for uncoated pellets are applicable to the ZrB_2 IFBA design. Gadolinia bearing burnable absorber fuel pellets are also an option in the AP1000 PWR fuel rod design. The inclusion of gadolinia in the fuel matrix acts as an impurity, and the unirradiated thermal conductivity of the Gd_2O_3-UO_2 fuel are appropriately reduced as a function of the gadolinia content. The staff agrees with the applicant's method and conclusion based on the conservative assumptions made in the applicant's models.

The fission product generation in Gd_2O_3-UO_2 fuel is similar to that for UO_2 fuel, and no further changes are required to appropriately assess TCD in gadolinia fuel. In summary, the PAD 4.0 TCD assessment tool is applicable for use with all fuel types considered in the AP1000 PWR core design. Based on the Halden test rods design as being applicable to the fuel pellet types used in the AP1000 PWR fuel rod design, the "effects of enrichment" as input parameter, the conservative modeling of the ZrB_2 coating, and the fission product generation in Gd_2O_3-UO_2, the staff concludes that the PAD 4.0 TCD assessment tool is suitable for use with all fuel types considered in the AP1000 PWR core design.

4.5. LOCA Analysis

4.5.1. SBLOCA

4.5.1.1. Thermal Conductivity Degradation

Reference 2 discusses the impact of TCD on the SBLOCA analyses for the AP1000 design. Westinghouse concludes that the increase in stored energy caused by TCD would not impact the conclusions in ~~Revision 19 of the AP1000 DCD~~the documented analyses since stored energy is not a significant effect in the SBLOCA transients. Westinghouse also mentions that variations in rod internal pressures relative to burnup are already covered to a large extent in the SBLOCA burnup studies that are performed as necessary with the NOTRUMP evaluation model ~~were performed in~~ (References 24 and 25). Westinghouse further states that burnup studies are not required specifically for the AP1000 since the SBLOCA analyses for the AP1000 plant design show that no core uncovery occurs, which means no clad heatup calculations are needed. Westinghouse explains that since clad heatup calculations are not performed, it would not be

Comment [c18]: This change is recommended because the TCD evaluation for SBLOCA in the cited Reference 2 (LTR-NRC-12-56) was developed for Revision 0 of the CRR. The impacts of TCD on the DCD SBLOCA results were evaluated separately and the estimated effect was provided via 50.46 reporting in DCP_NRC_003242.

Comment [c19]: This change is recommended for better consistency with Reference 2 because burnup studies are not contained in References 24 and 25; rather, those references detail the NOTRUMP (Appendix K) evaluation model.

necessary to perform burnup studies to look at the effects of TCD.

Reference 2 describes a quantitative analysis demonstrating that TCD has a negligible impact on the SBLOCA analysis of records performed with the Westinghouse NOTRUMP evaluation methodology for Westinghouse nuclear stream supply system (NSSS) plant designs. It is further stated that there is no need to credit additional margins to continue to meet the acceptance criteria. Westinghouse explains that ~~these~~the conclusions relative to the impact of TCD from the quantitative analyses are applicable to the AP1000.

Comment [c20]: Editorial clarification to better align with the discussion in Reference 2.

In RAI CRR-009-SI, the staff requested the applicant to analyze how much fuel rod heatup is calculated and whether this leads to significant changes to the previously calculated rod internal pressures during the SBLOCA when considering TCD effects. In the response to RAI-009-SI (Reference 13), Westinghouse states that during a SBLOCA, the fuel rods are surrounded by a two-phase mixture of steam and liquid water for the majority of the transient and that stored energy in the core because of the initial fuel temperature is not a significant effect in SBLOCA transients because the initial phase of the transient is gradual enough to remove the energy from the system before any core uncovery.

After the initial submittal of the CRR (Reference 1), and subsequent RAI response (Reference 13), Westinghouse identified the single failure of one automatic depressurization system stage 4 (ADS-4) valve on the nonpressurizer side of the reactor coolant system (RCS) to be more limiting than the analysis initially presented, with the failure on the pressurizer side. Westinghouse submitted the revised analysis in Revision 1 of the CRR (Reference 37) for the 2-inch (5 centimeter) cold leg break and inadvertent ADS transient simulations, in which containment backpressure was now used instead of atmospheric pressure during the transient. The revised analysis (Reference 37) shows core uncovery for the Inadvertent ADS and the 2-inch (5 centimeter) cold leg break cases. Since the analysis now shows that there is core uncovery, the analysis uses the SBLOCTA computer code to calculate the PCT during the transient. The SBLOCTA code is based on the LOCTA-IV code ~~approved~~ documented in WCAP-8301-P~~-A~~ (Reference 38), ~~and~~ modified as described in WCAP-10054-P-A and WCAP-10081-A (References 25 and 39), and approved with that application. The revised analyses still produces results that demonstrate significant margin to peak cladding temperature regulatory limits. Also, as a consequence of the new analysis showing that core uncovery occurs, the applicant assessed the effect of TCD on the revised analysis and concluded that the SBLOCA analysis is negligibly impacted (Reference 37). During an audit held on November 25, 2013 and November 26, 2013 (Reference 40) Westinghouse provided clarifying information on the impact of the revised analysis and containment backpressure methodology that is being used in the SBLOCA analysis. The applicant provided additional information to support that the containment pressure used was a conservatively low minimum containment pressure ~~(Reference 38)~~. Based on the information provided in References 37~~, 38,~~ and 40, the staff finds the new analysis, based on approved methodologies, demonstrates that the SBLOCA analysis results are negligibly impacted when considering the effects of TCD and that the revised analysis demonstrates compliance with the regulatory requirements.

Comment [c21]: Editorial update to clarify that the LOCTA-IV code report was not explicitly approved but that the SBLOCTA code was approved as part of the approval in WCAP-10054-P-A/WCAP-10081-A.

The editorial update also points to the references for both WCAP-10054-P-A and WCAP-10081-A.

Comment [c22]: Editorial markups because Reference 38 is the LOCTA-IV report (WCAP-8301) and does not provide information on the containment backpressure.

4.5.2. LBLOCA

4.5.2.1. Thermal Conductivity Degradation

Section 4.0 of Reference 10 discusses the impact of TCD on the LBLOCA analyses. Westinghouse explains how it estimated the effect of incorporating a fuel thermal conductivity model that considers fuel TCD as a function of burnup on ASTRUM (automated statistical treatment of uncertainty method). ASTRUM is documented in Reference 26, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," which does not explicitly account for fuel TCD. To account for fuel TCD, Westinghouse distinguishes between the "margin PCT" and the "integrated PCT." The margin PCT is established to serve as a reference PCT that better captures the true impact of the fuel TCD effects established by the integrated PCT. The margin PCT uses a reduced F_Q peaking factor conservatism only for the limiting two PCT cases from the original ASTRUM analysis where the conservatism was found to be attributed to an unreasonably ~~nonconservative~~ conservative F_Q rather than some real physical phenomena. The margin PCT is then established by finding the limiting PCT value among this updated original set of 124 WCOBRA/TRAC results. The limiting PCT was reduced by ~~21~~ 39 °C (70 °F) in this more realistic estimate (again, not assessing the impact of TCD).

Comment [c23]: Editorial change. Reference LTR-NRC-12-46. The margin PCT uses a reduced F_Q peaking factor conservatism because the F_Q values in the limiting two PCT cases were unreasonably conservative.

Comment [c24]: Editorial change. Correct conversion of delta PCT from °F to °C.

To establish the integrated PCT (or the PCT considering TCD effects in addition to adjustment of the F_Q conservatism in the original two limiting cases), 31 WCOBRA/TRAC cases were re-run, selected to be "among the most limiting cases from the original 124-run ASTRUM analysis." The overall process to quantify the effect of TCD in these 31 cases was to (1) update the previously approved PAD 4.0 fuel performance code to include a burnup-dependent fuel TCD model to initialize the WCOBRA/TRAC runs, (2) update the previously approved HOTSPOT code to use a burnup-dependent fuel TCD model (different from that used in PAD 4.0) throughout the extent of the transient, and (3) use the burnup-dependent peaking factors as given in Table 5-2 of Reference 10. Table 5-3 in Reference 10 shows that the design basis limiting PCT increased by ~~21~~ 38 °C (69 °F). Comparing the margin PCT to the integrated PCT, the effect of TCD seen would cause an approximate increase of ~~59~~ 77 °C (139 °F) between the updated analysis and the previous analysis.

Comment [c25]: Editorial change. Correct conversion of delta PCT from °F to °C.

Comment [c26]: Editorial change. Correct conversion of delta PCT from °F to °C.

In RAI CRR-004, the staff requested that Westinghouse explain the process of determining average assembly peaking factors and burnups with ASTRUM to ensure that the initial stored energy of these average fuel assemblies is not being underestimated. The staff also requested justification for limiting the burnup of the average assemblies to 30 GWD/MTU. In the response to RAI CRR-004 (Reference 6), Westinghouse explains that when applying ASTRUM to the AP1000 design analyses, values between [ ]a,c are produced during sampling average burnup. Sensitivity studies were performed for (1) the limiting ASTRUM case before the TCD analysis, and (2) the limiting ASTRUM case from the TCD analysis. In both sensitivity studies, the average burnup was fixed to approximately [ ]a,c to show the impact (at the much higher average burnup upper limit) of the increased stored energy on PCT for the limiting hot rod location. The results showed that "the increased temperature in the average rods does not substantially impact the global response

and had only a small impact on the calculated hot rod PCTs (Figures 3A and 3B of the RAI CRR-004 response (Reference 6))." The staff analyzed the figures presented in the response and confirmed that there was negligible impact on the maximum hot rod PCT. Therefore, the issue of potential underestimation of stored energy in the average assemblies is resolved.

4.5.2.2. Axial Power Shape Generation for WCOBRA/TRAC Cases

In RAI CRR-005, the staff asked Westinghouse to describe the process used to adjust the F_Q values in select cases closer to the desired value and to explain why this adjustment is appropriate with respect to how certain parameters are sampled within ASTRUM. The parameters in question are called PBOT and PMID, and are sampled (within a prescribed envelope based on a specific nuclear core design) when performing an ASTRUM analysis. ~~PBOT and PMID define values for inputs such as the peak linear heat rate (PLHR) and F_Q for a given WCOBRA/TRAC case and define the axial power shape for a given case~~PBOT and PMID, with inputs such as the peak linear rate (PLHR) and F_Q for a given WCOBRA/TRAC case, define the axial power shape for a given case. In the response to RAI CRR-005 (Reference 5), Westinghouse explains that the PSHAPE routine [

]a,c Since PLHR is required to be at the desired value—or greater (Westinghouse points to Table 12-6 in the ASTRUM topical report (Reference 26), which defines the criterion)—this would imply that F_Q is required to be at the desired value or greater. Therefore, reducing the F_Q value to the value corresponding to the desired value for the PLHR would be consistent with the as-approved ASTRUM. Westinghouse states: [

]a,c

Table 2 of the CRR-005 response (Reference 5) shows that the F_Q values used in the final margin case runs exceed the initially sampled F_Q values by the PSHAPE routine (only the top two PCT cases were re-run). It also shows that PLHR was also matched, as shown in Table 3 of the response. The staff finds that modification of the PSHAPE code (by manually modifying PBOT/PMID parameters) to achieve a more desirable F_Q does not cause a deviation from ASTRUM, as previously approved, and in this example is acceptable.

A more in-depth review of the PSHAPE code used to generate the axial power shapes for the WCOBRA/TRAC calculations revealed that Westinghouse internal procedures allow randomly sampled parameters to be changed or biased in certain circumstances within a certain prescribed tolerance. Specifically, this is the case for the sampled PBOT and PMID parameters where internal guidance allows for randomly sampled PBOT and PMID values to be changed by [ ]a,c to allow physical power shapes for WCOBRA/TRAC cases to be defined while also maintaining certain criteria, such as the PLHR (and hence F_Q) and integrated

Comment [c27]: Editorial change. Reference LTR-NRC-12-86. Response to CRR-005 states: [

]a,c

Comment [c28]: Editorial change. Reference LTR-NRC-12-86. Response to CRR-005 states: [

]a,c

hot rod power criteria as given in Table 12-6 of the ASTRUM methodology topical report (Reference 26).

While this practice conforms to the criteria in the documented ASTRUM topical report, it does not conform to the technical staff's understanding of Westinghouse's process for the random sampling of input parameters, which is not documented in the ASTRUM topical report. Consequently, the staff does not find the biasing of randomly sampled parameters as part of ASTRUM to be generally acceptable since it is implied that random samples are unbiased. However, in this instance, Westinghouse has demonstrated that the criteria in Table 12-6 of the ASTRUM topical report that was previously approved by the staff are satisfied, despite the biasing of certain random parameters, and despite the fact that the parameters were biased [ ]a,c set by internal Westinghouse procedures. Furthermore, any biasing should be limited so as not to significantly change the physical behavior of a given case, and should be justified for each case in which it is necessary to meet certain evaluation criteria, such as that given in Table 12-6 of the ASTRUM topical report (Reference 26).

4.5.2.3. Peaking Factor Burndown Credit

As discussed in Reference 10, Westinghouse is taking credit for peaking factor burndown concurrent with accounting for TCD effects. In RAI CRR-006, the staff asked Westinghouse to describe how the burnup-dependent peaking factors were determined and then implemented, in the LBLOCA analysis. In the response to RAI CRR-006 (Reference 6), Westinghouse explains that the process of determining the hot rod burnup, and $F_{\Delta H}/F_Q$ at the corresponding burnup, is [

]a,c Peaking factor limits were then defined by conservatively bounding the steady-state AP1000 peaking factors as shown in Figures 1 and 2 of the response to RAI CRR-006-S1 (Reference 13). This means that $F_{\Delta H}/F_Q$ is determined [ ]a,c. Westinghouse states that it developed the peaking factor curves [

]a,c which can be seen in Figures 1 and 2 of the response to RAI CRR-006-S1 (Reference 13).

Taking credit for peaking factor burndown for LBLOCA, the analyses are consistent with ASTRUM, since the maximum $F_{\Delta H}/F_Q$ (for any given time-in-cycle) is still being used, only with less margin being maintained between the defined limit and a given sampled peaking factor. Westinghouse also states that the reduced peaking factors "will be confirmed to be met [and not exceeded] during the future reload process, similar to other limits." Tables 1 and 2 of the CRR-006-S1 response (Reference 13) give peaking factor limits for the AP1000, which will be confirmed as part of the cycle design analysis. No changes to the plant Technical Specifications or Core Operating Limits Report will be made.~~which is part of the core operating~~

~~limits report that will be referenced by plant TS.~~ Since Westinghouse has demonstrated that the AP1000 LBLOCA analyses still conform to the as-approved ASTRUM, the staff finds taking credit for peaking factor burndown in the manner described above to be acceptable.

Comment [c29]: Basis for change: Response to CRR-006-S1 in LTR-NRC-13-18 states:

From a Technical Specification Surveillance perspective, this results in the lower burnup fuel to be more limiting, implicitly enveloping the higher burnup fuel. Therefore, no further Technical Specification Surveillance actions or COLR changes are required; the burndown credits supporting the LBLOCA PCT evaluations are confirmed analytically as part of the normal core design reload safety analyses process.

This approach is consistent with the application of burndown limits for all other LOCA analyses performed by Westinghouse with explicit accounting for TCD.

4.5.2.4. Summary of LBLOCA Analysis Changes in References 1 and 37

During an audit of the CRR (Reference 14), the staff reviewed a calculation note that led to questions about the assumptions used in the updated LBLOCA analyses that address the TCD concerns. There was a specific question regarding the type of reactor coolant pump (RCP) used. RAI CRR-007 was issued requesting clarification of the significant differences or changes to the LBLOCA analyses that address TCD, along with why it was necessary and appropriate for each change. In the response to RAI CRR-007(Reference 5), Westinghouse clarified that the RUV RCP designed by KSB was not used for DCD or CRR best estimate LOCA analyses, and the analyses in the calculation note in question are not applicable to the AP1000. Westinghouse goes on to state the primary differences between the DCD and CRR LBLOCA analyses, including the reasons for the changes, are as follows:

1. changes to the fuel assembly mechanical design features (primarily the grid design) because of the fuel design change

2. changes to the RCP design and homologous curves because of updated parameters resulting from design finalization

3. changes to the upper head structures because of updated parameters resulting from design finalization

4. increase in the time before RCP trip following an LBLOCA when offsite power is available for consistency with updated non-LOCA analyses for the CRR

5. change in PBOT or PMID box because of the core design change

6. change in $F_{\Delta H}$ limit because of the core design change

The response to RAI CRR-007 (Reference 5) also includes a notification of changes to ASTRUM to account for burnup-dependent aspects of the fuel performance changes since TCD and peaking factor burndown were not explicitly considered in the as-approved ASTRUM. These changes effectively include (1) expanding the fuel performance data to account for the effects of TCD, (2) [ ]a,c, (3) updating the thermal conductivity model in WCOBRA/TRAC and HOTSPOT, and (4) [ ]a,c.

Based on the above discussion, the staff finds that Westinghouse has incorporated all of the important design changes between the DCD and Reference 37 in the LBLOCA analyses.

The changes that were made resulted from changes as detailed in Reference 37 and described in Reference 10, and were found to be appropriate.

To properly address fuel TCD for the LBLOCA analysis, the staff requested, in RAI CRR-008, that Westinghouse re-analyze all 124 WCOBRA/TRAC cases, as required by the ASTRUM evaluation methodology (Reference 26), rather than the subset of cases that Westinghouse determined to be the most limiting. In the response to RAI CRR-008 (Reference 5), Westinghouse states that a plant-specific adaptation of ASTRUM was applied to consider the burnup-dependent aspects of the fuel performance changes considering TCD. All of the previous DCD cases were re-run using the same inputs, with a few exceptions. The exceptions occurred for the following reasons:

1. to account for TCD effects (this includes use of PAD 4.0 with TCD modeled to initialize the WCOBRA/TRAC and HOTSPOT fuel rod temperatures and a modified fuel thermal conductivity model in WCOBRA/TRAC and HOTSPOT)

2. to account for peaking factor burndown (only sampled average burnups greater than 30 GWD/MTU were affected)

3. to allow modification of PBOT/PMID parameters for ~~the two~~three limiting PCT cases i.e., F_Q margin was reduced for ~~the two~~three limiting PCT cases consistent with the PLHR and integrated hot rod power criteria in the originally approved ASTRUM evaluation methodology).

In the response, Westinghouse also notes that the burndown after 55 GWD/MTU is different than presented in Reference 10 and that the additional burndown towards the end of life better reflects the power and peaking limits expected for higher burnup assemblies. Additionally, [

]a,c

Since all previous LBLOCA cases were re-run, [
]a,c, the staff finds that Westinghouse performed the minimum number of cases as required by ASTRUM, and that the updated analysis is acceptable because the applicable regulations in SRP/NUREG-0800 Section 15.6.5 continue to be met.

The staff also asked about the impact of the corrected fuel TCD models on the peak containment pressure and temperature calculation in RAI CRR-023. In the response to RAI-CRR-023 (Reference 6), Westinghouse states that the current value for initial core stored energy used for input to the LBLOCA mass and energy release remains conservative when the effects of TCD are considered and, therefore, there is no change to the peak calculated containment pressure or temperature resulting from a LBLOCA. More details on the method used are given in the response to RAI CRR-026, and this response was reviewed in Section 4.5 of this document.

Comment [c30]: Basis for change:
Three cases were adjusted: the top two limiting PCT cases, and the 4th ranked case. This is consistent with response to CRR-005 in LTR-NRC-12-86:
Similar axial power shape modifications will be made for other cases as necessary [

]a,c

In LTR-NRC-14-20 (Reference 41), the applicant notified the staff of errors identified in the evaluation models for the LBLOCA analysis presented in Reference 37. Reference 41 contains a tally of all errors and modifications to the LBLOCA and SBLOCA evaluation models per the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." The cumulative estimated PCT impact on the LBLOCA analysis is ~~1.1~~19 °C (34 °F), which is less than the ~~10~~ 28 °C (50 °F) threshold specified in 10 CFR 50.46. Per Section 5.1 of Reference 37, any applicant referencing this topical report must account for any and all errors/modifications in the evaluation model and disposition them in the results presented in connection with the site-specific application of Reference 37. This would include the ~~1.1~~19 °C (34 °F) reported in Reference 41. The staff finds that this meets the requirements of 10 CFR 50.46.

Comment [c31]: Editorial change. Correct conversion of delta PCT from °F to °C.

Comment [c32]: Editorial change. Correct conversion of delta PCT from °F to °C.

Comment [c33]: Editorial change. Correct conversion of delta PCT from °F to °C.

4.6. Non-LOCA Analyses

Reference 37, Appendix B, Section 15.0, "Accident Analyses," presents the changes to the design bases for the AP1000 accident analyses. Several proposed changes in assumptions and design parameters affect several analyses and are discussed in this section. Staff evaluation of specific analyses is discussed later.

The CRR includes an initial reactor core power of 3,449.15 MWt, which assumes a calorimetric uncertainty of 1 percent, in analyses that previously assumed 3,483.3 MWt. This is consistent with the power uncertainty assumed for the large-break LOCA analysis. The difference of 15.15 MWt between non-LOCA and LOCA analyses is because of the reactor coolant pumps, which are not assumed to be operating during a LOCA because they trip automatically following an "S" signal. The applicant previously revised DCD Section 15.0.15 to include COL Information Item 15.0.15.1, which requires the COL holder to calculate the primary power calorimetric uncertainty before fuel load to confirm that the safety analysis primary power calorimetric uncertainty bounds the calculated value. Therefore, the staff found the use of 1 percent initial core power uncertainty acceptable since it continues to follow the guidance found in SRP/NUREG-0800 Section 15.

In RAI CRR-018, the staff requested Westinghouse to demonstrate that the in-containment equipment program has already taken into account the containment heatup events resulting from new procedures that discharge high temperature and high pressure primary coolant in the in-containment RWST. In its response (Reference 6), the applicant referred to DCD Appendix 3D for the equipment qualification methodology and stated that the aging program includes the effect of temperature for normal plant operation, plus 72 hours of accumulated Group 1 conditions and 24 hours of accumulated Group 2 conditions. The applicant stated that as long as these events do not exceed the temperatures and accumulated durations allowed for Group 1 or Group 2 events, then the affected components are bounded by component pre-accident aging design basis conditions. Based on the staff's review of the applicant's response (Reference 5), the staff found the response acceptable and considers RAI CRR-018 closed.

In RAI CRR-021, the staff requested that the applicant evaluate the difference between the FIGHTH code and the fuel performance code, PAD, which models the thermal conductivity degradation properly. The staff further requested that the applicant use this evaluation to determine the impact on the calculated DNBR, peak linear power density, transient power, power level, and cladding strain. In its response (Reference 5), the applicant stated that the PAD and FIGHTH results tend to agree very well, except for some specific instances. The differences between PAD and FIGHTH occur in the burnup range of 5 to 10 GWD/T and linear heat rates of 16.3 kW/ft or greater. The LHGR rate peaks at approximately 13.4 kW/ft, which is lower than the level at which differences are observed between PAD and FIGHTH. The applicant further stated that the PAD code is used for the safety analyses; therefore, the differences in the FIGHTH and PAD code will not have any impact on the calculation of DNBR, peak fuel temperatures, and enthalpy. Based the staff's review of the additional information provided by the applicant in Reference 5, the staff found the response acceptable and considers RAI CRR-021 closed.

In RAI CRR-026, the staff requested that the applicant describe the core initial stored energy calculation method and explain how zero rated power can be most limiting in regard to initial core stored energy. In its response (Reference 5), the applicant provided a description of the core stored energy calculation, which is normalized with respect to local linear power. Because of the normalization while the absolute core stored energy increases with regard to core power, the normalized core stored energy is limiting at low power (1.5 kW/ft), not zero power. The applicant provided analysis showing that when the effects of thermal conductivity degradation are included, the maximum normalized core stored energy is at a high burnup and is higher than the previously calculated normalized core stored energy. The applicant further stated that when using a more representative calculation that assumed a less conservative power distribution and a weighted average for burnup, this offset the increase in the normalized core stored energy by including the effects of thermal conductivity degradation. Based on the staff's review of the applicant's response (Reference 5), the staff found the response acceptable and considers RAI CRR-026 closed.

4.6.1. (SRP 15.1.1) Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature

The staff reviewed Section 15.1.1 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.1.1 through 15.1.4, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow."

The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- addition of the flow skirt

- increased lower core support plate flow hole size
- increased pressurizer volume
- increased reactor vessel (RV) diameter for the neutron pad addition
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The new analysis indicates that the decrease in feedwater flow temperature remains bounded by the analysis of excessive increase in secondary steam flow (Section 15.1.3) but is no longer bounded by the analysis for feedwater system malfunctions that result in an increase in feedwater flow (Section 15.1.2). The staff found these changes acceptable and confirms that the analysis of Section 15.1.3 continues to bound the analysis of Section 15.1.1.

4.6.2. (SRP 15.1.2) Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

The staff reviewed Section 15.1.2 in Appendix B Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis: increased $F_{\Delta H}$ limit (1.65 to 1.72), increased pressurizer volume, increased RV diameter for the neutron pad addition, increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients. The analysis was also updated and expanded to include flow increases to both steam generators. Because a primary coolant coastdown no longer occurs in the analysis, the FRACTRAN and VIPRE-01, which were previously used for this portion of the analysis, are no longer used.

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.1.1 through 15.1.4, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow," and found the changes acceptable and that all regulatory requirements continue to be met.

4.6.3. (SRP 15.1.3) Excessive Increase in Secondary Steam Flow

The staff reviewed Section 15.1.3 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- addition of the flow skirt

- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased main steam safety valve (MSSV) inlet piping diameter (increased 3 centimeters (1.2 inches))
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

Because a primary coolant coastdown no longer occurs in the analysis, the FRACTRAN and VIPRE-01 codes, which were previously used for this portion of the analysis, are no longer used.

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.1.1 through15.1.4, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow" and found the changes acceptable and that all regulatory requirements continue to be met.

4.6.4. (SRP 15.1.4) Inadvertent Opening of a Steam Generator Relief or Safety Valve

The staff reviewed Section 15.1.4 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- addition of the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increase MSSV inlet piping diameter (increased 3 centimeters (1.2 inches))
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.1.1 through 15.1.4, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve," and found the changes acceptable and that all regulatory requirements continue to be met.

4.6.5. (SRP 15.1.5) Steam System Piping Failure

The staff reviewed Section 15.1.5 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- addition of the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

In addition to the changes to the analysis, editorial changes were also included. In particular, the initial iodine and noble gas primary coolant concentrations are more accurately described as being based on their respective TS (i.e., equilibrium operating limits) because the TS limits do not necessarily correspond to the design fuel defect level. This is consistent with the modeling used in the analyses. The doses were revised based on updated analysis.

The applicant included in Reference 37 the analyses for steam system piping failures initiated from at power conditions that had been previously considered by the applicant. The results of analyses for steam system piping failures initiated from full power were demonstrated to meet the acceptance criteria.

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment," and found the changes acceptable and that all regulatory requirements continue to be met.

4.6.6. (SRP 15.1.1 and 15.1.4) Inadvertent Operation of the Passive Residual Heat Removal (PRHR) Heat Exchanger

The staff reviewed Section 15.1.6 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.1.1 and 15.1.4, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow."

This section was largely unchanged except for minor editorial changes. The staff accepts the editorial changes and agrees that no changes to the analysis were necessary based on the fuel design changes presented in Reference 37.

4.6.7. (SRP 15.2.1) Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow

No changes were made to this section. The staff agrees that no changes to the analysis were necessary based on the fuel design changes presented in Reference 37.

4.6.8. (SRP 15.2.2) Loss of External Electrical Load

Editorial changes were incorporated. The staff accepts the editorial changes and agrees that no changes to the analysis were necessary based on the fuel design changes presented in Reference 37.

4.6.9. (SRP 15.2.3) Turbine Trip

The staff reviewed Section 15.2.3 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- use of the digital ΔT signal
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients.
- modeled the moderator density function as a function of density

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.2.1 through 15.2.5, "Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of

Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)," and found the changes acceptable and that all regulatory requirements continue to be met.

4.6.10. (SRP 15.2.4) Inadvertent Closure of Main Steam Isolation Valves

No changes were made to this section. The staff agrees that no changes to the analysis were necessary based on the fuel design changes presented in Reference 37.

4.6.11. (SRP 15.2.5) Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip

No changes were made to this section. The staff agrees that no changes to the analysis were necessary based on the fuel design changes presented in Reference 37.

4.6.12. (SRP 15.2.6) Loss of Power to the Plant Auxiliaries

The staff reviewed Section 15.2.6 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- applied containment backpressure effects on PRHR heat transfer
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The loss of ac power to the plant auxiliaries case presented in the DCD, where feedwater flow is lost at time zero, and power to the reactor coolant pumps is lost as a result of the turbine trip, was renamed in Section 15.2.7 as "Loss of Normal Feedwater Flow with Loss of Offsite Power." The case presented in Section 15.2.6 now assumes a loss of reactor coolant pumps and loss of feedwater pumps at event initiation.

In RAI CRR-022, the staff requested quantitative evidence to justify the conclusion that the effect of TCD on loss of flow and loss of offsite power (LOOP) events would not exceed the 50 psid (pounds per square inch differential) margin of system pressure. In its response (Reference 5), the applicant provided results from its quantitative analysis showing sufficient margin to the reactor coolant system (RCS) Pressure Acceptance Criterion for both the loss of load and turbine trip event and the locked rotor event. For both events, the applicant showed that the increase in peak RCS pressure was less than 3 psid, which is small compared to the margin available in the analysis. Based on the quantitative analysis, the applicant provided the response (Reference 5). The staff finds the response to RAI CRR-022 acceptable and considers the matter closed.

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.2.6, "Loss of Nonemergency AC Power to the Station Auxiliaries." Based on the information provided in Reference 37 and the response to RAI CRR-022 (Reference 5), the staff finds the changes and analysis acceptable.

4.~~5~~6.13 (SRP 15.2.7) Loss of Normal Feedwater Flow

Comment [c34]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.2.7 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- containment backpressure effects on PRHR heat transfer
- addition of the flow skirt, increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased rod drop time for the safety analysis and the updated valve
- nozzle, and piping pressure loss coefficients

Editorial changes were made to the loss of feedwater analyses to identify an operator action to open the safety-related reactor vessel head vent to prevent filling the reactor coolant system water solid. An additional case, loss of normal feedwater flow with loss of offsite power, was added to this section.

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.2.7, "Loss of Normal Feedwater Flow," and found the changes acceptable and that the regulatory requirements continue to be met.

4.~~5~~6.14 (SRP 15.2.8) Feedwater System Pipe Break

Comment [c35]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.2.8 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)

- applied containment backpressure effects on PRHR heat transfer
- added the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.2.8, "Feedwater System Pipe Breaks Inside and Outside Containment," and found the changes acceptable and that the regulatory requirements continue to be met.

4.~~5~~6.15 (SRP 15.3.1) Partial Loss of Forced Reactor Coolant Flow

Comment [c36]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.3.1 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- added the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients.
- modeled the moderator density function as a function of density

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.3.1

through 15.3.2, "Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions," and found the changes acceptable and that the regulatory requirements continue to be met.

4.56.16 (SRP 15.3.2) Complete Loss of Forced Reactor Coolant Flow

Comment [c37]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.3.2 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- added the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.3.1 through 15.3.2, "Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions," and found the changes acceptable and that the regulatory requirements continue to be met.

4.56.17 (SRP 15.3.3) Reactor Coolant Pump Shaft Seizure (Locked Rotor)

Comment [c38]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.3.3 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- added the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume

- increased RV diameter for the neutron pad addition
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients.
- modeled the moderator density function as a function of density

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.3.3 through 15.3.4, "Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break," and SRP 15.0.3, "Design Basis Accident Radiological and Consequence Analyses for Advanced Light Water Reactors." The staff found the changes acceptable and that the regulations continue to be met.

4.~~5~~6.18 (SRP 15.3.4) Reactor Coolant Pump Shaft Break

Comment [c39]: Editorial correction. Break in numerical sequence for sub-section 4.6.

Only editorial changes were incorporated. The staff accepts the editorial changes and agrees that no changes to the analysis are necessary based on the fuel design changes presented in Reference 37.

4.~~5~~6.19 (SRP 15.4.1) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition

Comment [c40]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.4.1 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.4.1, "Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition," and found the changes acceptable and that the regulations continue to be met.

4.~~5~~6.20 (SRP 15.4.2) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

Comment [c41]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.4.2 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- use of the digital ΔT signal
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," and found the changes acceptable and that the regulations continue to be met.

4.~~5~~6.21 (SRP 15.4.3) Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error)

Comment [c42]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.4.3 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.4.3, "Control Rod Misoperation (System Malfunction or Operator Error," and found the changes acceptable and that the regulations continue to be met.

4.~~5~~6.22 (SRP 15.4.4) Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

Comment [c43]: Editorial correction. Break in numerical sequence for sub-section 4.6.

There are no proposed changes to this section. The staff agrees that no changes to the analysis were necessary based on the fuel design changes presented in Reference 37.

4.~~5~~6.23 (SRP 15.4.5) A Malfunction or Failure of the Flow Controller in a Boiling-Water Reactor Loop that Results in an Increased Reactor Coolant Flow Rate

Comment [c44]: Editorial correction. Break in numerical sequence for sub-section 4.6.

There are no proposed changes to this section. The staff agrees that no changes to the analysis were necessary based on the fuel design changes presented in Reference 37.

4.~~5~~6.24 (SRP 15.4.6) Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant

Comment [c45]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.4.6 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- added the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- use of the digital ΔT signal
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.4.6, "Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)" and found the changes acceptable and that the regulations continue to be met.

4.~~5~~6.25 (SRP 15.4.7) Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

Comment [c46]: Editorial correction. Break in numerical sequence for sub-section 4.6.

There are no proposed changes to this section. The staff agrees that no changes to the analysis were necessary based on the fuel design changes presented in Reference 37.

4.~~5~~6.26 (SRP 15.4.8) Spectrum of Rod Cluster Control Assembly Ejection Accidents

Comment [c47]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.4.8 in Appendix B of Reference 1, which addresses changes and updates to the spectrum of rod ejection accidents described in Revision 19 of the AP1000 DCD. In RAI CRR-019, the staff noted that the rod ejection accident was analyzed using the previous fuel pellet thermal conductivity model and asked what the quantitative impact of the corrected TCD model with burnup dependence will be on the rod ejection accident analysis. In its response dated January 2, 2013, the applicant stated that the effects of TCD were already taken into account in the nuclear power transient in the calculation of the fuel temperatures used for Doppler feedback effects and for the maximum heat flux and minimum DNBR portion of the hot rod transient response analysis, which, [ ]a,c. For the minimum heat flux and maximum fuel temperature and enthalpy portion of the hot rod transient response, the applicant provided quantitative analysis showing that the primary effect of TCD on the CRR rod ejection peak fuel temperatures and enthalpies will be in the [ ]a,c. Furthermore, the applicant provided quantitative analysis results that show that the change in enthalpy rise is more than offset by the margin calculated for the CRR to the criterion in SRP 4.2 Revision 3 and that the peak fuel centerline temperature also remains well below the fuel melt temperature. Based on the staff's review of the applicant's response dated January 2, 2013, the staff accepts the response and RAI CRR-~~020~~ 019 is considered closed.

Comment [c48]: Changed to be consistent with the RAI requesting information related to TCD: CRR-019 (consistent with 2nd sentence of 1st paragraph).

In RAI CRR-020, the staff noted that a 3-D kinetics code is used to analyze the AP1000 rod ejection accident and asked if the same code will be used for the future reload analysis. In its response dated January 2, 2013, the applicant stated that the same 3-D kinetics methodology used in Reference 1 rod ejection analysis will be used for reload cycles. The applicant further stated that the methodology is consistent with Reference 27 and that benchmark testing discussed in RAI CRR-003 demonstrated that the 3-D kinetics code used to perform the calculations gives equivalent results to SPNOVA for representative rod ejection accidents. Based on the staff's review of the applicant's response dated January 2, 2013, the staff accepts the response and RAI CRR-020 is considered closed.

During the Phase 2 audit (Reference 14), several rod ejection analyses were reviewed, including the hot zero power (HZP) enthalpy analysis [ ]a,c, the hot full power (HFP) enthalpy analysis [ ]a,c, and the DNBR analysis were reviewed during the audit. All cases were analyzed for both the first cycle and the equilibrium cycle.

Westinghouse identified the HZP analysis as limiting for the PCMI and the zero power high clad temperature failure criteria. The corresponding acceptance criterion is less than 170 cal/g (or less than 150 cal/g if rod internal pressure exceeds RCS pressure) to mitigate high clad temperature failure. The PCMI failure criterion compares the peak fuel average enthalpy rise versus the cladding oxide to wall thickness ratio. Westinghouse states that the PCMI failure criterion is converted to a rod average burnup basis for comparison directly to ANC. For the HZP [ ]a,c scenario, peak enthalpy rise values were ~~was~~ calculated for the first cycle and the reference 18-month equilibrium cycle, [ ]a,c cal/g and [ ]a,c cal/g, respectively, which ~~is~~ are well below the PCMI limit. A figure in the corresponding Westinghouse calculation note comparing the PCMI failure criterion to the calculated

Comment [c49]: Text added since there are two high clad temperature criteria; one based on peak enthalpy for cases originated from HZP and one based on DNB for cases originated from >5% power.

Comment [c50]: Text added because PCMI is enthalpy rise, not peak enthalpy.

Comment [c51]: Changed these values to reflect the enthalpy rise values calculated.

Comment [c52]: Editorial change

Comment [c53]: Added this text to clarify which limit these values correspond to, since both PCMI and HZP high clad temp are discussed in the same paragraph.

enthalpy rise results of the re-analysis as a function of rod average burnup also was reviewed, along with another figure showing the calculated enthalpy rise margin. It was noted that the minimum margin is [ ]a,c percent or [ ]a,c cal/g. The calculated peak enthalpy results of [ ]a,c cal/g and [ ]a,c cal/g for the first cycle and the reference 18-month equilibrium cycle were also found to be well below the high clad temperature failure criterion.

Comment [c54]: Added this sentence to report the peak enthalpy values used to compare to the high clad temperature limits for cases initiated from HZP conditions.

Note that these values are consistent with the above enthalpy rise values and differ from those values by the initial enthalpy at HZP conditions (17.5 cal/g).

Westinghouse identified the HFP analysis as limiting for the core coolability criterion, which requires the peak enthalpy to be less than 230 cal/g. Westinghouse states that it has traditionally assumed the peak enthalpy limit to be 200 cal/g. ~~In the corresponding Westinghouse calculation note reviewed, two scoping cases were performed looking at conservative, but unrealistic, xenon skewing to obtain initial condition F_Q peaking factors close to the TS limit. The two cases correspond to the reference 18-month equilibrium cycle for the DCD and the first cycle for the DCD. The results of the two cases were [ ]a,c for the equilibrium cycle and [ ]a,c for the first cycle. It is stated that for the final case for the CRR, a bounding but more feasible approach is taken.~~ The analysis results show a peak enthalpy of [ ]a,c cal/g ([ ]a,c °C or [ ]a,c °F with [ ]a,c percent melt), which is significantly less than the recommended SRP 4.2 limit of 230 cal/g, and also less than the 200 cal/g internal limit imposed by Westinghouse.

Comment [c55]: This block of text has been deleted because:

1)It mentions that the fuel management is from the DCD which is not correct. The Cycle 1 model is consistent with the AFC described in Ref. 37 and the EQCY is the model used for other safety analyses performed for Ref. 37.

2) [ ]a,c

The final analysis reviewed by the staff addresses the changes made to the rod ejection DNBR analyses based on the core design changes documented in the CRR. The corresponding calculation note reviewed showed calculation results for the first cycle and reference 18-month equilibrium cycle. The acceptance criterion, as dictated by SRP 4.2, for high cladding temperature failure by DNB, is that the number of failed rods cannot exceed that number assumed in the dose analysis. For the AP1000, this number corresponds to less than or equal to [ ]a,c percent of the rods in the core. The safety analysis limit for DNBR is [ ]a,c for the AP1000. Westinghouse assumes that [ ]a,c with a hot rod DNBR less than [ ]a,c will fail, and are counted toward the [ ]a,c percent rod failure allowance, which is noted to be conservative because [ ]a,c. The results based on the updated analysis show that the first cycle failure rate is [ ]a,c percent and the equilibrium cycle is [ ]a,c percent, both of which meet the [ ]a,c percent limit with significant margin.

Comment [c56]: [ ]a,c

The staff conducted its evaluation in accordance with the guidelines provided in SRP Section 15.4.8, Revision 3, "Spectrum of Rod Ejection Accidents (PWR)," SRP Section 4.2, Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents," and SRP 15.0.3, "Design Basis Accident Radiological and Consequence Analyses for Advanced Light Water Reactors." The staff concluded that the calculations reviewed during the audit supported the applicant's analysis presented in Reference 37. Additionally, the staff finds that the applicant follows the guidance provided in SRP Sections 15.4.8 and 4.2, Appendix B. Therefore, the staff finds the analysis to be acceptable.

4.~~5~~6.27 (SRP 15.5.1) Inadvertent Operation of the CMT during Power Operation

Comment [c57]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.5.1 in Appendix B of Reference 37, which addresses changes

and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- added the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition, containment backpressure effects on PRHR heat transfer
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

Editorial changes were made to the inadvertent core makeup tank (CMT) analyses to identify an operator action to open the safety-related reactor vessel head vent to prevent filling the reactor coolant system water solid.

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.5.1 through 15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," and found the changes acceptable and that the regulations continue to be met.

4.~~5~~6.28 (SRP 15.5.2) CVS Malfunction that Increases Reactor Coolant Inventory

Comment [c58]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.5.2 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- added the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition, containment backpressure effects on PRHR heat transfer
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

Editorial changes were made to the inadvertent chemical and volume control analyses to identify an operator action to open the safety-related reactor vessel head vent to prevent filling

the reactor coolant system water solid.

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Sections 15.5.1 through 15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," and found the changes acceptable and that the regulations continue to be met.

4.~~5~~6.29 (SRP 15.6.1) Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS

Comment [c59]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.6.1 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- added the flow skirt
- increased lower core support plate flow hole size
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- use of the digital ΔT signal
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

As stated in 10 CFR Part 50, GDC 17, "Electric Power Systems," analysis of coincident loss of ac power for an RCS depressurization event is not required based on the turbine and RCP response to this scenario. With a loss of ac power, the OTΔT is the trip signal. The low pressurizer pressure signal is the actuated protection signal. Since the applicant no longer assumes a loss of offsite power and accompanying primary coolant coastdown in the analysis, the FRACTRAN and VIPRE-01 which were previously used for this portion of the analysis, are no longer used. The staff finds this to be acceptable since it conforms with the new analysis methodology and provides greater clarity.

4.~~5~~6.30 (SRP 15.6.2) Failure of Small Lines Carrying Primary Coolant Outside Containment

Comment [c60]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.6.2 in Appendix B of Reference 37, which addresses changes

and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The applicant updated the description of the initial iodine and noble gas primary coolant concentrations as based on their respective TS (i.e., equilibrium operating limits) because

the TS limits do not necessarily correspond to the design fuel defect level. This is consistent with the modeling used in the analyses.

The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- use of the digital ΔT signal
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.6.2, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve," and SRP 15.0.3, "Design Basis Accident Radiological and Consequence Analyses for Advanced Light Water Reactors." The staff found the changes acceptable and that the regulations continue to be met.

4.~~5~~6.31 (SRP 15.6.3) Steam Generator Tube Rupture

Comment [c61]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.6.3 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The applicant revised the analysis to incorporate updates to the NSSS model and also incorporate the resolution to the containment backpressure issue. The following changes were incorporated in the updated analysis:

- increased $F_{\Delta H}$ limit (1.65 to 1.72)
- increased pressurizer volume
- increased RV diameter for the neutron pad addition
- increased MSSV inlet piping diameter (increased 3 centimeters or 1.2 inches)
- increased rod drop time for the safety analysis and the updated valve, nozzle, and piping pressure loss coefficients

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure," and SRP 15.0.3, "Design Basis Accident Radiological and Consequence Analyses for Advanced Light Water Reactors." The staff found the changes acceptable and that the regulations continue to be met.

4.~~5~~6.32 (SRP 15.7.4) Radioactive Release from a Subsystem or Component

Comment [c62]: Editorial correction. Break in numerical sequence for sub-section 4.6.

The staff reviewed Section 15.7.4 in Appendix B of Reference 37, which addresses changes and updates to the accident analyses described in Revision 19 of the AP1000 DCD. The following changes were incorporated in the updated analysis:

- used a 1 percent power measurement uncertainty
- revised the core source terms based on the AFC
- increased the radial peaking factor from 1.65 to 1.75

The changes in the analysis reflect the design information provided in Reference 37. The staff conducted the review in accordance with the guidelines provided in SRP Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," and SRP 15.0.3, "Design Basis Accident Radiological and Consequence Analyses for Advanced Light Water Reactors." The staff found the changes acceptable and that the regulations continue to be met.

4.~~5~~6.33 (SRP 15.8) Anticipated Transients without Scram

Comment [c63]: Editorial correction. Break in numerical sequence for sub-section 4.6.

There are no proposed changes in this section. The staff agrees that no changes to the analysis were necessary based on the fuel design changes presented in Reference 37.

5.0 STAFF CONCLUSIONS

The staff has completed its review of the AP1000 CRR—as described in Reference 1 and supplemented by Reference 2, Reference 3, Reference 10, Reference 37, and Reference 41—and concludes that the applicant has demonstrated that the improvements to the fuel, core components, and core design continue to comply with the regulatory requirements by following the guidance provided in the respective SRP sections, with respect to the AP1000 standard plant design. The staff's conclusions for specific technical topics are found within the respective technical evaluation sections of this report. The staff notes that per Section 5.1 of Reference 37, any applicant referencing this topical report is required to include all impacts of changes to the methods and models to demonstrate acceptable results for the ECCS. This includes all changes to the methods and models since the submittal of revised LOCA analyses, beginning with the impacts noted in Reference 41.

The staff does not generally approve biasing of randomly sampled parameters. However, as detailed in Section 4.~~4~~5.2.2 of this safety evaluation, the staff concludes that the applicant demonstrated compliance with criteria previously approved by the staff.

Comment [c64]: Editorial revision. Biasing of randomly sampled parameters is discussed in Section 4.5.2.2; Section 4.4.2.2 does not exist in this report.

The staff, therefore, approves the use of the AP1000 CRR (Reference 37) to be referenced by AP1000 COL license holders.

6.0 Conditions and Limitations

The staff's approval of this topical report is specific to the AP1000 generic design. Any use in whole or in part for other designs would require additional applicability review by the staff.

7.0 References

1. WCAP-17524-P, Revision 0, "AP1000 Core Reference Report," March 2012 (Agencywide Documents Access and Management System (ADAMS) No. ML120750528).

2. LTR-NRC-12-56, "Supplemental Information to WCAP-17524, "AP1000 Core Reference Report' To Address the Impact of Thermal Conductivity Degradation on Additional Events," August 21, 2012, (ADAMS No. ML12251A259~~ML12242A288~~).

3. LTR-NRC-13-26, "Supplemental Information on End-of-Life Seismic/LOCA Calculations for the AP1000 Pressurized Water Reactor," April 30, 2013, (ADAMS Nos. ML13128A018 and ML13128A017).

4. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," March 2007 (ADAMS No. ML070810350).

5. LTR-NRC-12-86, "Westinghouse Response to NRC RAIs on WCAP-17524, "AP1000 Core Reference Report," January 2, 2013 (ADAMS Nos. ML130080344 and ML13008A267).

6. LTR-NRC-13-3, "Second Transmittal of Westinghouse Responses to NRC RAIs on WCAP-17524, "AP1000 Core Reference Repot," January 10, 2013 (ADAMS Nos. ML13022A106 and ML13022A105).

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8. WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," Revision 2, October 2011 (ADAMS No. ML11292A021).

9. WCAP-15466-P, Revision 2, "AP1000 Code Applicability Report," March 2004 (ADAMS Nos. ML040890149 and ML032671338).

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Comment [c65]: Editorial correction, the incorrect ADAMS number was originally listed

Comment [c66]: WCAP-16943-P-A should be listed to be consistent with the citation of this reference in Section 4.2.1.

Comment [c67]: Editorial correction. The incorrect ADAMS number was originally listed.

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25. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985 (ADAMS No. ML100050586).

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Comment [c68]: Editorial clarification for consistency with the markups in Section 4.5.1.1.

Comment [c69]: Editorial update to match usage in Section 4.5.1.1.

Comment [c70]: Missing Reference cited in Section 4.3