Table of Contents

List of Tables

List of Figures

15.0 Accident Analyses

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15.1 Methodology

15.1.1 Overview

This chapter details the expected response of the plant to the spectrum of transients and accidents which constitute the design basis events. The methodologies used to analyze the Chapter 15 transients and accidents fall into three general categories. These are the non-LOCA transient and accident analysis methodologies which are detailed in the Duke Power topical report DPC-NE-3005-PA (Reference 1), the AREVA NP. LOCA analysis methodology (Reference 2) described in Section 15.14, and the Duke Power offsite dose analysis methodology described in Section 15.1.10.

The DPC-NE-3005-PA topical report methodology was used to establish a new set of licensing basis analyses beginning with Oconee Unit 2 Cycle 18. The following transients and accidents are analyzed with the new methodology. The specific cases analyzed for each transient or accident are listed in Table 15-32.

15.2 Startup Accident

15.3 Rod Withdrawal at Power Accident

15.4 Moderator Dilution Accidents

15.5 Cold Water Accident

15.6 Loss of Coolant Flow Accidents

15.7 Control Rod Misalignment Accidents

15.8 Turbine Trip Accident

15.9 Steam Generator Tube Rupture Accident

15.12 Rod Ejection Accident

15.13 Steam Line Break Accident

15.17 Small Steam Line Break Accident

Section 15.1, "Uncompensated Operating Reactivity Changes", in the original FSAR was deleted since the plant transient response due to the effects of fuel depletion and xenon buildup are insignificant and do not challenge the Reactor Protective and Engineered Safeguards Systems or approach any design limits. Sections 15.10, 15.11, 15.15, and 15.16 do not require thermal-hydraulic transient analyses methods and were not reanalyzed in DPC-NE-3005-PA.

15.1.2 Topical Reports

The topical reports which describe the analysis methodologies used in this chapter are as follows:

DPC-NE-3000-PA

DPC-NE-3000-PA, "Thermal-Hydraulic Transient Analysis Methodology," (Reference 4) describes the RETRAN-3D (Reference 38) system transient thermal-hydraulic models and the VIPRE-01 (Reference 6) core transient thermal-hydraulic models used by Duke Power to analyze most of the non-LOCA transients and accidents. This report includes the standard nodalization model and the various code options that are used.

DPC-NE-3005-PA

DPC-NE-3005-PA, "UFSAR Chapter 15 Transient Analysis Methodology," (Reference 1) describes the Duke Power methodology for analyzing the UFSAR Chapter 15 non-LOCA transients and accidents for the Oconee Nuclear Station. This report includes a description of the computer codes used, the physics parameters, the setpoint methodology, and details of the initial conditions, boundary conditions, acceptance criteria, and all other aspects of the methodology. The computer codes comprising this methodology are RETRAN-3D (Reference 38), VIPRE-01 (Reference 6), CASMO-3 (Reference 7) or CASMO-4 (Reference 44), SIMULATE-3 (Reference 8), SIMULATE- 3K (Reference 9), and TACO-3 (Reference 10).

DPC-NE-1004-A

DPC-NE-1004-A, "Nuclear Design Methodology Using CASMO-3 / SIMULATE-3P," (Reference 11) describes the Duke Power methodology for the neutronic simulation of the Oconee reactors with the CASMO-3 (Reference 7)/ SIMULATE-3 (Reference 8) codes.

DPC-NE-1006-PA

DPC-NE-1006-PA, “Oconee Nuclear Design Methodology using CASMO-4/SIMULATE-3” (Reference 45) describes the Duke Power methodology for the neutronic simulation of the Oconee reactors with the CASMO-4 (Reference 44)/ SIMULATE-3 (Reference 8) codes.

DPC-NE-2003-PA

DPC-NE-2003-PA, "Core Thermal-Hydraulic Methodology Using VIPRE-01," (Reference 12) describes the Duke Power methodology for core thermal-hydraulic analysis for Oconee using the VIPRE-01 code. The non-statistical DNBR limit using the BWU CHF correlation is developed in this report.

DPC-NE-2005-PA

DPC-NE-2005-PA, "Thermal-Hydraulic Statistical Core Design Methodology," (Reference 13) describes the Duke Power methodology for determining the statistical DNBR limits using the VIPRE-01 code. This methodology allows the uncertainty in many of the DNB-related parameters to be combined into a statistical DNBR limit, rather than to include each uncertainty explicitly in the thermal-hydraulic analysis. For some of the transients and accidents the primary flowrate associated with less than four pumps in operation, and the higher flow uncertainty at reduced flowrates, result in different statistical DNBR design limits. The applicable limit is given for each analysis. The non-statistical DNBR limits using the BWU correlations are developed in this report.

BAW-10192-PA

BAW-10192-PA, "BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," (Reference 2) describes the RELAP5-based AREVA NP, LOCA Evaluation Model. This topical report has been accepted by the NRC as in compliance with 10 CFR Appendix K (Reference 14). The model changes necessary to analyze M5 cladding are contained in a separate topical report (Reference 37). The computer codes which comprise this methodology are RELAP5/MOD2-B&W (Reference 15), CONTEMPT (Reference 16), REFLOD3B (Reference 17), and BEACH (Reference 18). The Oconee large-break and small-break LOCA events are analyzed with this Evaluation Model.

DPC-NE-2015

DPC-NE-2015, “Oconee Nuclear Station Mark-B-HTP Fuel Transition Methodology,” (Reference 40) describes the methodology to be used by Duke Energy Carolina, LLC (Duke) for performing the core reload design, the fuel assembly mechanical and thermal-hydraulic analyses, and the

UFSAR Chapter 15 non-LOCA transient and accident analyses, for the transition to the AREVA NP Mark-B-HTP fuel assembly design at the Oconee Nuclear Station. Included in this report are the methods to evaluate the mixed-core effects of the Mark-B-HTP fuel design with the current Mark-B11 fuel design. These methods are presented as revisions to Duke's existing methodology reports that have been previously approved by the NRC. The reports relative to UFSAR Section 15.1 are: DPC-NE-2003P-A, DPC-NE-2005P-A, DPC-NE-3000-PA, and DPC-NE-3005-PA. The methodology revisions also include changes that are not associated with the change in the fuel assembly design. These changes are enhancements to the existing methods to improve analytical margins, to correct errors, and to provide editorial clarification. A brief summary of the AREVA NP methods for performing the LOCA analyses consistent with the requirements of 10 CFR 50.46 and Appendix K is also presented.

Paragraph(s) Deleted Per 2000 Update

15.1.3 Computer Codes and CHF Correlations

RETRAN-3D

The non-LOCA system transient thermal-hydraulic analyses use the RETRAN-3D code (Reference 38). RETRAN-3D was developed by Computer Simulation & Analysis, Inc. for EPRI to enhance and extend the simulation capabilities of the RETRAN-02 code (Reference 5). RETRAN-O2 has the flexibility to model any general fluid system by partitioning the system into a one-dimensional network of fluid volumes and connecting junctions. The mass, momentum, and energy equations are then solved by employing a semi-implicit solution method. The equations are based on a homogeneous two-phase mixture, with capability for phase separation via bubble rise and slip models. A non-equilibrium pressurizer model, special component models for pumps, valves, and control systems, and general heat transfer modeling are included.

RETRAN-3D has many new and enhanced capabilities relative to RETRAN-02, in particular, a 3-D kinetics core model, improved two-phase models, an improved heat transfer correlation package, and an implicit numerical solution method. Most of the capabilities of the RETRAN-02 code have been retained within RETRAN-3D as options, except for a limited number of models and correlations that were not in use. For transients which challenge the DNBR limit, RETRAN-3D provides core boundary conditions to VIPRE-01 and SIMULATE-3.

VIPRE-01

The core thermal-hydraulic and fuel pin analyses use the VIPRE-01 code (Reference 6) developed by the Electric Power Research Institute. VIPRE-01 uses the subchannel analysis approach in which the fuel assembly is divided into a number of quasi-one-dimensional channels that communicate laterally by diversion crossflow and turbulent mixing. Conservation equations of mass, axial and lateral flow, and momentum are solved. The flow field is assumed to be incompressible and homogeneous, with models for subcooled boiling and co-current phase slip. VIPRE-01 accepts boundary conditions from RETRAN-3D and SIMULATE-3 and determines the DNBR using the applicable CHF correlations.

CASMO-3

Nuclear constants are generated with the Studsvik of America code CASMO-3 (Reference 7) for use in Oconee reload design (Reference 11). CASMO-3 is used for generating data used as input to the SIMULATE codes.

CASMO-4

Nuclear constants are generated with Studsvik Scandpower code CASMO-4 (Reference 44) for use in Oconee reload design (Reference 45). CASMO-4 is used for generating data used as input to the SIMULATE codes.

SIMULATE-3

Nuclear parameters and core power distributions are generated with the Studsvik of America code SIMULATE-3 (Reference 8) for use in Oconee reload design (References 11 and 45). Nuclear constants are input to SIMULATE-3 from the CASMO-3 or CASMO-4 code. SIMULATE-3 outputs are input to the RETRAN-3D and VIPRE-01 codes.

SIMULATE-3K

The Studsvik of America code SIMULATE-3K (Reference 9) is used for transient three-dimensional modeling of the rod ejection accident. SIMULATE-3K provides the same neutronics solution to steady-state 3-D calculations as SIMULATE-3. Nuclear constants are input to SIMULATE-3 from the CASMO-3 or CASMO-4 code. SIMULATE-3K rod ejection analysis results are input to RETRAN-3D and VIPRE- 01.

Deleted Paragraph(s) per 2008 Update

TACO-3

The TACO-3 code (Reference 10) developed by AREVA NP is used to calculate the initial fuel pin thermal and mechanical conditions for the non-LOCA analyses performed by Duke Power, and for the LOCA analyses performed by AREVA NP.

RELAP5/MOD2-B&W

The RELAP5/MOD2-B&W code (Reference 15) developed by AREVA NP, is used for best-estimate and licensing transient simulation of pressurized water reactors. It has also been modified to include the conservative models required for LOCA analysis per Appendix K to 10 CFR 50 (Reference 14). The solution technique contains two energy equations, a two-step numerics option, a gap conductance model, constitutive models, and control and component system models. This code is used for the blowdown simulation in Oconee large-break LOCA analyses, and for the thermal-hydraulic response in the small-break LOCA analyses.

CONTEMPT

The CONTEMPT code (Reference 16) as modified by AREVA NP, is used to calculate the containment pressure following LOCA. The containment pressure is used as an input to the RELAP5 blowdown analysis and the REFLOD3 refill and reflood analysis.

REFLOD3B

The REFLOD3B code (Reference 17) developed by AREVA NP, is used for simulation of the refill and reflood periods of the large-break LOCA analysis. The program calculates flows, mass and energy inventories, pressures, temperatures, and steam qualities along with variables associated with the refilling of the reactor lower plenum and the recovery of the core.

BEACH

The BEACH code (Reference 18) developed by AREVA NP, is used for the prediction of reflood heat transfer during the large-break LOCA analysis. It calculates the peak cladding temperature and the local oxidation for comparison with the 10 CFR 50.46 (Reference 22) acceptance criteria.

Paragraph(s) Deleted Per 2000 Update

BHTP Critical Heat Flux Correlation

The BHTP critical heat flux correlation (Reference 41) is used in the VIPRE-01 code to calculate the DNBR for non-LOCA transient and accident analyses for the Mark-B-HTP fuel assembly design.

BWU-Z Critical Heat Flux Correlation

The BWU-Z critical heat flux correlation (Reference 24) is used in the VIPRE-01 code to calculate the DNBR for non-LOCA transient and accident analyses for fuel assemblies with mixing vane grids.

BWU-N Critical Heat Flux Correlation

The BWU-N critical heat flux correlation (Reference 24) is used in the VIPRE-01 code to calculate the DNBR for non-LOCA transient and accident analyses for fuel assemblies with mixing vane grids, but in the lower part of the fuel assembly where there are no mixing vane grids. This correlation can also be used for the steam line break DNBR analysis.

W-3S Critical Heat Flux Correlation

The W-3S critical heat flux correlation as programmed in the VIPRE-01 code (Reference 6) is used to calculate the DNBR for the steam line break accident, when the core conditions are beyond the correlation ranges for the other critical heat flux correlations.

Modified-Barnett CHF Correlation

The modified-Barnett (MBAR) CHF correlation (Reference 42) is used to calculate the DNBR for the steam line break accident for the Mark-B-HTP fuel assembly design when the core conditions are beyond the BHTP correlation ranges.

15.1.4 Initial Conditions

The generic initial conditions assumed in the transient and accident analyses are summarized in Table 15-34 and referenced figures. These values have been selected to ensure that the results of each analysis have an appropriate level of overall conservatism. Many of the initial conditions are determined based on the nominal value of the plant parameter plus or minus the uncertainty associated with each parameter. Parameters for which the uncertainty is included in the statistical DNBR limit are set to the nominal value. Initial conditions which are not included in this table are provided in the detailed description of each analysis.

Sometimes it is desirable to extend the full power operation of a reload cycle by reducing the average Reactor Coolant System temperature (RCS T-ave) at end-of-cycle (EOC) conditions. Reducing RCS T-ave adds positive reactivity due to moderator temperature feedback and extends the full power operation capabilities. The safety analyses events described in this chapter have been evaluated for an end-of-cycle T-ave reduction of up to 10°F lower than the RCS T-ave values shown in Table 15-34. The 10°F reduced RCS T-ave with 4 Reactor Coolant Pumps (RCPs) operating is acceptable and does not create more limiting accident results than those reported in this chapter.

15.1.5 Setpoints and Delay Times

The Reactor Protective System and Engineered Safeguards Protective System trip setpoints and delay times are summarized in Table 15-35. The setpoints are based on the technical specification values, and are either increased or decreased to account for setpoint drift depending on whether an earlier or later reactor trip is conservative. Trip delay times account for instrument string delays and component delays, such as the control rod gripper coil release delay.

15.1.6 Reactivity Insertion Following Reactor Trip

The reactivity insertion following reactor trip is a combination of a minimum available tripped rod worth and a normalized insertion rate. The minimum available tripped rod worth assumed in safety analyses must ensure, as a minimum, that the shutdown margin in the technical specifications is preserved. This shutdown margin assumes that the most reactive rod remains in the fully withdrawn position and that the other control rods drop from their power dependent insertion limits. The normalized reactivity insertion rate is determined by bounding control rod drop times as determined by plant testing, and by developing a conservative relationship between rod position and normalized reactivity worth.

15.1.7 Decay Heat

In the non-LOCA transients and accident analyses for which the post-trip decay heat is an important modeling consideration, the ANSI/ANS-5.1-1979 Standard (Reference 25) is used. The inputs to the calculation of the time-dependent decay heat per the ANS Standard are based on Oconee-specific core physics parameters. This modeling is implemented in the application of the RETRAN-3D code using either the built-in ANS standard with inputs to account for Oconee-specific core parameters, or as an input table of decay heat vs. time. The decay heat modeled by AREVA NP in the LOCA analysis is 1.2 times the 1971 ANS Standard as required by 10 CFR 50 Appendix K (Reference 14).

15.1.8 Single Failure and Loss of Offsite Power Assumptions

A limiting active single failure in the Reactor Protective System or in the Engineered Safeguards is assumed. A single failure in the Emergency Feedwater System is also considered. A failure of the manual atmospheric dump valves is not considered. A loss of offsite power is only applied to the Section 15.13 steam line break accident, for which it is assumed to be lost at time zero, and for the Section 15.14 LOCA analyses.

15.1.9 Credit for Control Systems and Non-Safety Components and Systems

Control systems are generally assumed to respond as designed or remain in manual control (inactive), whichever assumption is more conservative. Non-safety components and systems are generally not credited in the analyses. The following are specific exceptions to the general modeling philosophy on control systems, and the situations where non-safety components and systems are credited in the analyses:

1. In the dropped rod event, the Integrated Control System will respond by initiating a plant runback to a reduced power level. Since this plant runback assists in the mitigation of the dropped rod event, no credit is taken for this control system design feature. This assumption is an additional conservatism that is not required by the methodology philosophy.

2. For a loss of all reactor coolant pumps without a loss of the Main Feedwater System, the Integrated Control System is credited for raising steam generator levels to the natural circulation setpoint. This design feature is implicitly credited in the loss of coolant flow event, and involves non-safety equipment. A failure of this design function would be mitigated manually by operator action to start the Emergency Feedwater (EFW) System.

3. The moderator dilution accident credits the control rod insertion limit alarm to alert the operator that a boron dilution event is in progress. This alarm relies on non-safety equipment and the plant computer.

4. Many of the transient and accident analyses involve control rod movement. These analyses credit the normal withdrawal sequence, overlap, and rod speed, which are controlled by non-safety control systems.
5. For certain failures in the EFW System, credit is taken for realigning EFW flow through the non-safety MFW System.
6. Steaming of the steam generators with manual non-safety atmospheric dump valves is credited.
7. Deleted per 2003 update
8. The capability to remotely throttle certain valves is credited. Some of the controls required to remotely throttle these valves are not safety-grade.
9. Electrical bus voltage and frequency control are credited. These are controlled by non-safety components.
10. The Integrated Control System trips both main feedwater pumps on a high steam generator level indication. A high level indication may occur following a main steam line break due to the pressure drops that result from the blowdown of the steam generator. Tripping of the main feedwater pumps will be assumed to occur in the steam line break analysis only if the plant response is more limiting.

15.1.10 Environmental Consequences Calculation Methodology

Environmental Consequences

A summary of the offsite doses is presented in Table 15-16. A description of each accident analysis is given in the appropriate section.

Fission Product Inventories

Inventory in the Core: Fission product inventories within the core are calculated based on the ORIGEN methodology (e.g., ORIGEN-ARP or SAS2H/ORIGEN-S of the SCALE computer code)(Section 15.1, Ref. 27). The core inventories for the Maximum Hypothetical Accident are shown in Table 15-15.

Inventory in the Fuel Pellet Clad Gap: The fuel pin gap activities were determined using Regulatory Guide 1.183 (Section 15.1, Ref. 35). For non-DNB fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals (Reference 46 and 47). A maximum of 25 fuel rods, per fuel assembly, shall be allowed to exceed the rod power/burnup criteria for Footnote 11 in RG 1.183 in accordance with the license amendment request submitted by letter dated July 15, 2015 (Reference 46). The fuel cycle design ensures that none of these fuel pins experience DNB following any design basis accident. The environmental consequences of the control rod ejection accident, and fuel handing accidents are based on the assumption that the fission products in the gap between the fuel pellets and the cladding of the damaged fuel rods are released as a result of cladding failure. The inventories used for the control rod cluster assembly ejection accident are shown in Table 15-50. The gap inventory for the fuel handling accident is shown in Table 15-1.

Inventory in the Reactor Coolant: The quantity of fission products released to the reactor coolant during steady state operation is based on the use of escape rate coefficients (sec^{-1}) derived from experiments involving purposely defected fuel elements. (Section 15.1, References 29, 30, 31, 32) These coefficients represent the fraction of the activity in the fuel

that is released, per unit time. Values of the escape rate coefficients used in the calculations are shown in Table 11-4.

Calculations of isotopic specific activities in the reactor coolant arising from steady-state fission product releases from the fuel (except for Kr-85) were performed with the Duke computer code PWR-SOURCE. The code calculates equilibrium reactor coolant fission product inventories and specific activities from the steady-state solutions to the differential equations for the radioactive decay chains for more than 150 isotopes. Due to the extremely long half life of Kr-85, an equilibrium activity level will not be reached in the reactor coolant during an operating cycle. For this particular isotope, the activity level is calculated from the exact solution of the decay chain, utilizing equilibrium activities of parent isotopes as inputs.

The reactor coolant activity levels are listed in Table 15-51. Dose Equivalent Iodine (DEI) and Dose Equivalent Xenon (DEX) calculations are shown in Table 15-65 and Table 15-66.

Inventory in the OTSGs and Secondary-Side Systems: The concentration of the iodine isotopes in the steam generators and secondary system coolant are assumed to be at the Technical Specification limit of 0.1 μCi/gm dose equivalent I-131, unless otherwise stated in a specific accident analysis. No credit is taken for removal of iodine from the secondary coolant by station demineralizers.

The concentrations of noble gases in the secondary side coolant are assumed to be negligible, and therefore are not modeled. Noble gases entering the secondary coolant system are continuously vented to the atmosphere via the condenser off-gas system. Thus, there would be only very small quantities of these gases within the secondary side coolant that could be released during an accident, and their contribution to the overall whole body dose will be negligible.

Calculation of Accident Doses

The Code of Federal Regulations, Title 10, Part 100, Section 11 (Section 15.1, Ref. 34) requires a dose consequence evaluation of postulated accidents resulting in fission product releases to the environment. Two types of doses are calculated for purposes of analyzing these accidents: internal doses to the thyroid resulting from inhalation of iodines and external whole body doses resulting from submersion in noble gases and iodines.

The dose consequences of a Maximum Hypothetical Accident, a Rod Ejection Accident, Large and Small Main Steam Line Break Accidents and Fuel Handling Accidents have been evaluated using an Alternative Source Term in accordance with the Code of Federal Regulations, Title 10, Part 50, Section 67 (Reference 39). For these evaluations, a total effective dose equivalent (TEDE) dose is calculated. Control room doses are also reported for these accidents.

Doses are calculated at two locations: the exclusion area boundary (EAB) and the outer boundary of the low population zone (LPZ). Doses calculated at the EAB and LPZ are modeled as a receptor located in a semi-infinite cloud of activity per Reg. Guide 1.109. (Section 15.1, Ref. 33).

For accidents using Alternative Source Term methodology, control room doses are calculated, and follow Regulatory Guide 1.183 (Reference 35). Values assumed for rate of unfiltered inleakage into the control room and airflow imbalance between dual control room air intakes bound the tested site values.

Atmospheric dispersion factors (χ/Qs) used in calculating control room doses are given in Table 15-61, and conform in general to the regulatory positions of Regulatory Guide 1.194 (Reference 43). Table 15-61 values represent bounding χ/Qs from a particular release type to either control

room air intake. For use in dose analyses, these values may be adjusted to represent Oconee's dual control room intake configuration.

15.1.11 Reload Safety Evaluation

Each fuel reload cycle design is reviewed to determine if the values of the safety analysis physics parameters assumed in the UFSAR Chapter 15 licensing basis transient and accident analyses remain valid. If the licensing basis assumptions remain bounding for the reload core, then no additional actions are required. If the predicted values violate the licensing basis assumptions for any of the key parameters, then reanalysis of the affected transients and accidents is required.

15.1.12 Use of Westinghouse WH-177 Lead Test Assemblies

Technical Specification 4.2.1 allows for a limited number of lead test assemblies (LTAs) to be included in the reactor core. As required in this technical specification, these LTAs are placed in non-limiting locations. Although currently there are no LTAs in use, previous Oconee core designs have used LTAs with the Westinghouse WH-177 fuel design. These LTAs have some differences in thermal-hydraulic parameters due to variations in the assembly design relative to the Framatome Mk-B11 and Mk-B10 fuel designs comprising the rest of the core. These design differences are described in UFSAR Section 4.2.2.2.1.

The Westinghouse WH-177 LTAs were evaluated with respect to the transients and accidents contained in Chapter 15 of the UFSAR and appropriate analyses were performed. Chapter 15 of the UFSAR contains transients and accidents that are sensitive to global and local effects. Global analyses whose results are controlled by core average parameters are not affected by the presence of LTAs. The core transient analysis for any of the non-LOCA design basis transients or accidents that are potentially sensitive to local effects were explicitly analyzed for the differences in hydraulic design and performance of the different fuel assembly types. An evaluation was also performed for the LOCA analysis.

The behavior of the minimum departure from nucleate boiling ratio (DNBR) was calculated for the mixed core of LTAs and Framatome Mk-B11 and Mk-B10 fuels. The co-resident fuel types were analyzed with their respective critical heat flux correlations and limits. As a result each fuel type has specific limits that include the effects of flow variations as well as fuel assembly feature performance. The limits derived from these calculations were applied to the LTAs and the Mk-B11 and Mk-B10 fuels to ensure DNBR criterion was met.

Centerline fuel melt (CFM) checks were performed for both the Framatome Mk-B11 and Mk-B10 fuels and the Westinghouse WH-177 LTAs to ensure that the CFM criterion was met.

The REA peak fuel pellet enthalpy for the WH-177 LTAs is bounded by the Mark B fuel results due to the lower enrichment of the WH-177 LTAs. The analysis performed for the Mark B fuel demonstrated that the peak fuel enthalpy was well below the peak enthalpy acceptance criterion.

The LOCA analysis was also evaluated for WH-177 LTAs. Westinghouse determined a peaking penalty to ensure the WH-177 fuel is non-limiting. This peaking penalty was applied to the WH-177 LTAs when designing the Oconee Unit 3 core to assure that the LTAs are non-limiting with respect to LOCA acceptance criteria. Thus, the Framatome fuel assemblies remain the limiting fuel with respect to LOCA.

Deleted paragraph(s) per 2011 update.

15.1.13 USE of AREVA Mark-B-HTP Fuel Assemblies

Starting from Oconee Unit 1 Cycle 28, Oconee has transitioned to full cores of AREVA Mark-B-HTP fuel assemblies from mixed cores of AREVA Mark-B11 and AREVA Mark-B-HTP fuel assemblies. DPC-NE-2015-PA (Reference 40) describes the methodologies to be used by Duke for performing the UFSAR Chapter 15 non-LOCA transient and accident analyses, for the AREVA Mark-B-HTP fuel assemblies.

The AREVA Mark-B-HTP fuel assemblies are evaluated with respect to the transients and accidents contained in Chapter 15 of the UFSAR and appropriate analyses were performed. An evaluation was also performed for the LOCA analysis.

The behavior of the minimum departure from nucleate boiling ratio (DNBR) was calculated for the full core of Mark-B-HTP fuel. The limit derived from this calculation was applied to the Mk-B-HTP fuel to ensure DNBR criterion was met.

Centerline fuel melt (CFM) checks were performed for the Mark-B-HTP fuel to ensure that the CFM criterion was met. The REA peak fuel enthalpy was well below the peak enthalpy acceptance criteria for the Mark-B-HTP fuel.

The LOCA analysis was also evaluated for Mark-B-HTP fuel assemblies using the existing LOCA methodologies. All of the 10 CFR 50.46 criteria were met.

The full cores of Mark-B-HTP fuel assemblies contained in the Oconee cores were evaluated with respect to the transients and accidents contained in Chapter 15 of the UFSAR and found to meet all acceptance criteria.

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15.2 Startup Accident

15.2.1 Identification of Cause and Description

The startup accident is an uncontrolled withdrawal of a control rod group from a zero power initial condition. It is caused by an operator error or a malfunction in the Rod Control System and can result in a nuclear power excursion. Since the heat removal capability of the secondary system is not increased during the power excursion, the resultant power mismatch would cause an increase in the Reactor Coolant System (RCS) and secondary system temperatures and pressures. The control rod motion would also cause the core power peaking to change. The reactor would be expected to trip on high flux or high RCS pressure.

The startup accident is analyzed from a hot zero power beginning-of-cycle condition, with three reactor coolant pumps (RCPs) in operation. The maximum control rod withdrawal rate is assumed. The system analysis determines the transient peak RCS pressure, and the transient core boundary conditions for the detailed core thermal-hydraulic analysis. In the peak RCS pressure analysis, the pressurizer spray and the pressurizer PORV are assumed to be inoperable. The pressurizer code safety valves (PSVs) are modeled using conservative assumptions for drift, blowdown, and valve capacity that minimize relief flow. The analysis methodology and the computer codes used in the analysis are given in Table 15-33. The initial conditions are given in Table 15-34. The Reactor Protective System and Engineered Safeguards Protective System setpoints and delay times are given in Table 15-35.

The reactivity addition rate assumed in the analysis is based on control rod group overlap, rod speed, and withdrawal sequence, which are controlled by non-safety systems. The loop with two RCPs in operation will indicate a lower hot leg pressure than the loop with only one active RCP. Therefore, the analysis assumes a single failure of one of the narrow range pressure channels on the loop with only one active RCP. This requires the high pressure reactor trip to be generated by the loop with a lower RCS pressure, which is conservative since it will delay reactor trip.

The startup accident is considered to be a fault of moderate frequency. The acceptance criteria for this accident are that the peak RCS pressure does not exceed 110% (2750 psig) of the design pressure, and that the minimum DNBR remains above the design limit.

15.2.2 Analysis

The startup accident analysis assumes three RCPs in operation and considers a maximum control rod withdrawal rate of 11.5 pcm/sec. The system thermal-hydraulic analyses have been performed for a core loaded with Mk-B-HTP fuel. The results presented model the replacement steam generators. The analysis duration of 100 seconds is sufficient to demonstrate the peak thermal power and peak RCS pressure. The analysis results are shown in Figure 15-1 through Figure 15-6, and the sequence of events is given in Table 15-36. Figure 15-1 shows the neutron power and thermal power transients. Neutron power does not begin to appreciably increase until the inserted reactivity begins to approach one dollar at approximately 46 seconds. Reactor trip occurs on high power at 51.0 seconds with neutron power at approximately 155% of 2568 mwth. The thermal power rises to a peak value of 80.5% of 2568 mwth at 51.1 seconds. Figure 15-2 shows the reactivity response. The reactivity insertion rate due to rod withdrawal is constant until reactor trip. Fuel heatup causes negative reactivity insertion due to Doppler temperature feedback until reactor trip. System heatup prior to reactor trip causes the moderator temperature to increase, which inserts positive reactivity due to the assumed positive moderator temperature coefficient of reactivity. Figure 15-3 and Figure 15-4 show the cold leg

and hot leg temperature transients. Because of the reduced flow due to the inactive RCP, the temperature response in the loop with the inactive RCP is delayed. After reactor trip and the opening of the PSVs, the temperatures in both loops decrease. Figure 15-5 shows the pressurizer level response. During the thermal power excursion, level rises rapidly due to the insurge of liquid into the pressurizer. After reactor trip and the opening of the PSVs, the pressurizer level rises more slowly and then stabilizes. Figure 15-6 shows the RCS pressure (hot leg indication) as a function of time. RCS pressure rises to a maximum value of approximately 2673.5 psig at 53.8 seconds, and then decreases due to PSV lift. The peak RCS pressure of 2723.6 psig occurs at the bottom of the reactor vessel.

15.2.3 Conclusions

The startup accident results in a peak core thermal power of 80.5% of 2568 mwth. The RCS conditions at the peak thermal power, specifically core inlet flow and temperature, have significant margin to conditions leading to DNB. The cooler inlet flow and relatively high RCS flow provide additional DNB margin. Therefore, DNB is not a concern for this transient. The peak RCS pressure for this transient is 2723.6 psig. All of the acceptance criteria are met.

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15.3 Rod Withdrawal At Power Accident

15.3.1 Identification of Causes and Description

The rod withdrawal at power accident is caused by an operator error or a failure in the Rod Control System which results in an uncontrolled withdrawal of a control rod group while the reactor is at power. The rod withdrawal causes a nuclear power excursion and a resultant heatup and pressurization of the Reactor Coolant System (RCS). The expected plant response to a rod withdrawal event would include the following. Feedwater flow would follow the increase in reactor power, thereby maintaining adequate RCS heat removal until the reactor is tripped on high flux or flux/flow/imbalance. Following reactor trip, the Turbine Bypass System (TBS) and main steam code safety valves would relieve steam in order to control the post-trip steam generator pressures. RCS pressure would be controlled by the pressurizer spray, PORV, and heaters. In addition, feedwater would be automatically controlled to maintain the post-trip steam generator level.

Separate analyses are performed to investigate the peak RCS pressure and the core cooling capability following the rod withdrawal event. The results presented model the replacement steam generators. The core cooling analysis covers a spectrum of initial power levels that bounds the range of permissible power levels given the number of operating reactor coolant pumps (RCPs). Four and three RCPs in operation are considered. Initial power levels below 15% are assumed to be bounded by the startup accident. In the peak RCS pressure analysis, the pressurizer spray, pressurizer PORV, and the Turbine Bypass System are assumed to be inoperable. In addition, the pressurizer and main steam code safety valves are modeled using conservative assumptions for drift, blowdown, and valve capacity that minimize relief flow. Both the peak RCS pressure and the core cooling analyses hold main feedwater and main steam flow rates constant prior to reactor trip. The analysis methodology and the computer codes used in this analysis are given in Table 15-33. The initial conditions are given in Table 15-34. The Reactor Protective System and Engineered Safeguards System setpoints and delay times are given in Table 15-35.

The reactivity addition rates assumed in the analyses are bounded by minimum and maximum values which are calculated based on control rod group overlap, rod speed, and withdrawal sequence, which are controlled by non-safety systems. No single failure has been identified which adversely impacts the results of the cases initiated from four RCP operation. For the cases initiated from three RCP operation, the analysis assumes a single failure of one of the narrow range pressure channels on the loop with only one active RCP. This requires the high pressure reactor trip to be generated by the loop with a lower RCS pressure, which is conservative since it will delay reactor trip.

The rod withdrawal at power accident is considered to be a fault of moderate frequency. The acceptance criteria for this accident are that the minimum DNBR remains above the design limit, and the peak RCS pressure does not exceed 110% (2750 psig) of design pressure.

15.3.2 Peak RCS Pressure Analysis

The RETRAN system thermal-hydraulic analysis results are valid for the full core with Mk-B-HTP fuel.

The limiting peak RCS pressure case assumes a full power initial condition and a withdrawal rate equivalent to 2.5 pcm/sec. Since the maximum RCS pressure is expected to occur near the time of reactor trip, the analysis duration is 10 seconds following the reactor trip. The

transient response for this limiting case is shown in Figure 15-11, Figure 15-12, Figure 15-13, and Figure 15-14 and the sequence of events is given in Table 15-37. Neutron power (Figure 15-11) increases at a constant rate until the reactor trips on high RCS pressure at about 37 seconds. Since the reactivity insertion is fairly slow, the thermal power essentially stays in equilibrium with the neutron power prior to reactor trip. RCS hot and cold leg temperatures are given in Figure 15-12. The cold leg temperature increases gradually prior to trip and then increases rapidly following the turbine trip due to increasing saturation temperature in the steam generators. Hot leg temperatures increase both due to the rising cold leg temperatures and due to the increasing reactor power. Pressurizer level (Figure 15-13) increases steadily as the RCS heats up, expands, and causes an insurge into the pressurizer. The RCS pressure response (Figure 15-14) essentially mirrors the pressurizer level, with a peak value reached at about 41 seconds. At this point, a peak pressure of 2635.5 psig is reached at the bottom of the reactor vessel.

15.3.3 Core Cooling Capability Analysis

The RETRAN system thermal-hydraulic analysis results are valid for the full core with Mk-B-HTP fuel.

The limiting DNBR case assumes a full power initial condition and a withdrawal rate equivalent to 0.5 pcm/sec. The transient response for this limiting case is shown in Figure 15-15, Figure 15-16, Figure 15-17, Figure 15-113, and Figure 15-114, and the sequence of events is given in Table 15-38. While the trends are very similar to those shown in the peak RCS pressure case, the duration of the analysis is much longer due to a significantly lower reactivity insertion rate. Since the minimum DNBR occurs near the time of reactor trip, the analysis duration is 10 seconds following the reactor trip. In order to evaluate the transient DNBR, the system analysis results are input to a detailed core thermal-hydraulic analysis. Neutron power and thermal power (Figure 15-15) increase at a constant rate until the reactor trips on high RCS temperature at about 204 seconds. RCS hot and cold leg temperatures are given in Figure 15-16. The cold leg temperature increases gradually prior to trip and then increases rapidly following the turbine trip due to increasing saturation temperature in the steam generators. Hot leg temperatures increase both due to the rising cold leg temperatures and due to the increasing reactor power. Pressurizer level (Figure 15-17) increases steadily as the RCS heats up, expands, and causes an insurge into the pressurizer. The RCS pressure response (Figure 15-113) essentially mirrors the pressurizer level, although the increase is suppressed by pressurizer spray.

15.3.4 Conclusions

The rod withdrawal at power accident results in a peak RCS pressure of 2635.5 psig. The transient minimum DNBR (Figure 15-114) is 1.519 for the full core with Mk-B-HTP fuel at 205.2 seconds. The minimum DNBR value is above the design limit. All of the acceptance criteria are met.

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15.4 Moderator Dilution Accidents

15.4.1 Identification of Causes and Description

A moderator dilution accident occurs when the soluble boric acid concentration of makeup water supplied to the Reactor Coolant System (RCS) is less than the concentration of the existing reactor coolant, and the water is injected in an uncontrolled manner. The cause of such an event can be attributed to any one of a number of failure modes in the systems that are capable of supplying unborated water to the RCS. With the reactor initially at power, control rods would insert to offset the reduction in RCS boron concentration. The operator would be alerted by the control rod insertion and terminate the event by identifying the dilution source and isolating it.

The moderator dilution accident is analyzed at the initial conditions of beginning-of-cycle power operation (Mode 1) with the Integrated Control System (ICS) in either the automatic or manual mode. Manual operator action is relied on to terminate the dilution. Mode 1 is analyzed to demonstrate that there is adequate time for the operator to terminate the dilution when maximum dilution source flowrates are assumed. The accident is precluded in Mode 6 by Technical Specification 3.9.7. Therefore, no analysis is presented. In Mode 1 with the ICS in manual, mitigation does not begin until reactor trip occurs. This conservatively ignores any other alarms or indications of the increase in reactor power, pressurizer level, and RCS pressure. In Mode 1 with the ICS in automatic, mitigation of the event does not begin until the rod withdrawal limit alarm actuates. This conservatively ignores the indications of the control rods inserting to control the power level and temperature. The analysis assumes conservatively high dilution flowrates, high initial boron concentrations, and small mixing volumes. The moderator dilution accident potentially results in a loss of shutdown margin and an inadvertent criticality, approaching the DNBR limit, or challenging the peak RCS pressure limit. This accident is conservatively analyzed to ensure that the operator terminates the boron dilution prior to exceeding these criteria.

As discussed in the preceding paragraph, alarm actuation is credited for alerting the operator that a boron dilution event is in progress. The rod withdrawal limit alarm relies on non-safety equipment. No single failure has been identified that would prevent the operators from successfully isolating the possible dilution sources and terminating the accident.

The moderator dilution accident is considered to be a fault of moderate frequency. The acceptance criteria for manual operator action to terminate the dilution event is 15 minutes during Mode 1 following the actuation of the alarm credited for alerting the operator of the event. By meeting this operator action time and preventing core re-criticality, it is assured that the plant response will not approach the DNBR limit or the peak RCS pressure limit.

15.4.2 Full Power Initial Condition Analysis

<u>Mode 1 With ICS in Automatic</u>

A conservative upper bound on the dilution flowrate of 300 gpm of unborated water is assumed, which is the design capacity of two bleed transfer pumps. At this flowrate re-criticality would not occur until 17.2 minutes following the rod withdrawal limit alarm which alerts the operator. Therefore there is sufficient time for the operator to terminate the dilution event.

<u>Mode 1 With ICS in Manual</u>

A conservative upper bound on the dilution flowrate of 300 gpm of unborated water is assumed, which is the design capacity of two bleed transfer pumps. At this flowrate re-criticality would not

occur until 15.6 minutes following the reactor trip alarm which alerts the operator. Therefore there is sufficient time for the operator to terminate the dilution event.

15.4.3 Refueling Initial Condition Analysis

Technical Specification 3.9.7 isolates all unborated water sources in Mode 6. Therefore, this is not a credible accident in Mode 6.

15.4.4 Conclusions

Two moderator dilution accident cases were performed corresponding to Mode 1 with the ICS in automatic and Mode 1 with the ICS in manual. The Mode 1 analyses calculate 17.2 minute and 15.6 minute operator action times for the ICS in automatic and manual cases, respectively. The accident is not credible in Mode 6. All of the acceptance criteria are met.

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15.5 Cold Water Accident

15.5.1 Identification of Causes and Description

The cold water accident is caused by an inadvertent startup of the fourth reactor coolant pump (RCP) from an initial three RCP operating condition. The increase in core flow as a result of the fourth RCP starting causes a decrease in the core average temperature. If the moderator temperature coefficient of reactivity is negative, an insertion of positive reactivity and an increase in reactor power will occur. Administrative controls limit the power level at which the fourth RCP can be started to less than 50% power. The normal plant response to this event would be for the Integrated Control system (ICS) to insert control rods in an attempt to maintain the initial power level.

The cold water accident is analyzed from an 80% of 2568 MWth power end-of-cycle initial condition. A conservative RCP start time is assumed. The system analysis determines the transient core boundary conditions for the detailed core thermal-hydraulic analysis. It is assumed that rod control is in manual and the pressurizer heaters are inoperable. The pump control circuitry interlock that prevents startup of an idle pump if the power is above 50 percent full power is assumed to be inoperable. The analysis methodology and the computer codes used in this analysis are given in Table 15-33. The initial conditions are given in Table 15-34. The Reactor Protective System and Engineered Safeguards Protective System setpoints and delay times are given in Table 15-35.

No single failure has been identified which adversely affects this accident.

The cold water accident is considered to be a fault of moderate frequency. The acceptance criteria for this accident are that the minimum DNBR remains above the design limit, and the peak RCS pressure does not exceed 110% (2750 psig) of design pressure. Since this event results in a minor RCS pressurization that does not approach the limit, only the minimum DNBR acceptance criterion is of concern.

15.5.2 Analysis

The cold water accident analysis results are shown in Figure 15-18, Figure 15-115, Figure 15-116, Figure 15-117 and Figure 15-118 and the sequence of events is given in Table 15-39. The system thermal-hydraulic analyses were performed for the replacement steam generators with a core loaded with Mk-B-HTP fuel. Since the minimum DNBR occurs near the time the RCP has come up to speed, the analysis is terminated 10 seconds after the RCP achieves full speed. Following the start of the fourth RCP, RCS flow (Figure 15-18) rapidly increases to full flow, resulting in a decrease in the core average temperature (Figure 15-115). Neutron power and thermal power (Figure 15-116) increase during this time period due to the positive reactivity insertion from the decrease in the core average temperature, and reach maximum values of 107.6% and 97.5%, respectively. No reactor trip setpoints are exceeded. A combination of Doppler feedback and increasing RCS cold leg temperatures (Figure 15-117) after the pump has reached full speed stop the power excursion, with power nearly returning to its initial condition by the end of the analysis. The RCS pressure (Figure 15-118) does not go above 2200 psig during the simulation. Since the maximum thermal power that occurs during this event is less than 100% full power, and the other core conditions are relatively close to nominal full power conditions, DNB is not a concern during this event.

15.5.3 Conclusions

The results of the cold water accident demonstrate that since the maximum power level remains less than 100%, the minimum DNBR remains well above the limit. The RCS pressure transient does not approach the peak RCS pressure limit. All of the acceptance criteria are met.

15.5.4 References

1. Deleted per 1996 Update
2. Deleted per 1999 Update
3. Deleted per 1999 Update

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15.6 Loss of Coolant Flow Accidents

15.6.1 Identification of Cause and Description

A loss of coolant flow accident occurs if one or more of the reactor coolant pumps (RCPs) stops due to a loss of electrical power or a mechanical failure. The loss of coolant flow accident resulting from an electrical failure results in one or more RCPs coasting down. The limiting loss of coolant flow accident resulting from a mechanical failure is a locked rotor in one pump. If the reactor is at power at the time of the accident, the immediate effect of a loss of coolant flow is a rapid increase in the core coolant temperature. This temperature increase could result in approaching DNB with subsequent fuel damage if the reactor is not tripped promptly. During the loss of coolant flow accident, the Reactor Protective System (RPS) will trip the reactor on the flux/flow/imbalance trip, or on the pump monitor trip. If all RCPs trip, the plant transitions to the natural circulation mode of core cooling.

During a RCP coastdown event, the flux/flow/imbalance trip function trips the reactor when the setpoint is reached, and the pump monitor trip trips the reactor when any two of the four RCPs trip if the reactor power is greater than 2%. The pump monitor trip function has only one channel per pump. Therefore, assuming a single failure of the pump monitor trip on one pump, the possible RCP coastdown events with four or three RCPs in operation are determined. In order to evaluate the transient DNBR, the system analysis results are input to a detailed core thermal-hydraulic analysis. Since some of the RCP coastdown events are bounded by others, only the following five RCP coastdown events are analyzed. Results for Cases 2, 3, and 4 are presented since they bound the other cases.

Case	RCP Coastdown(1)	Power Level (%)	Trip Function
1	4/1	100	flux/flow
2	4/2(2)	100	flux/flow
3	4/4	100	pump monitor
4	3/1(2)	80	flux/flow
5	3/3	80	pump monitor

Note:

1. 4/1 means 1 RCP coasting down with 4 RCPs in operation
2. The RCP(s) coasting down can be in the same loop or in different loops

For the locked rotor accident analysis a single failure in the pump monitor trip is assumed for both four and three RCPs in operation. Therefore, the flux/flow/imbalance trip provides DNB protection for the locked rotor event. With three RCPs in operation, a locked rotor in the loop with both RCPs operating is the limiting case. In order to evaluate the transient DNBR, the system analysis results are input to a detailed core thermal-hydraulic analysis. The results presented model the replacement steam generators.

The analysis methodology and the computer codes used in the loss of flow accident analyses are given in Table 15-33. The initial conditions are given in Table 15-34. Beginning-of-cycle conditions are limiting. The RPS and Engineered Safeguards Protective System setpoints and delay times are given in Table 15-35.

A single failure in the pump monitor trip function is assumed in the loss of flow accident analyses. This failure results in relying on the flux/flow/imbalance trip function to trip the reactor in most of the analyzed cases. The RCS will transition to the natural circulation cooling mode if all RCPs have stopped. Natural circulation is then established by raising steam generator levels to the natural circulation setpoint. If the Main Feedwater System is in operation, the increase in steam generator levels is controlled by the non-safety Integrated Control System. Otherwise, the Emergency Feedwater System actuates and the safety-grade Emergency Feedwater Control System controls the steam generator level to the natural circulation setpoint.

The RCP coastdown accidents are considered to be faults of moderate frequency (fewer than all RCPs coast down) or infrequent fault (all RCPs coast down) events. The acceptance criterion for all RCP coastdown accidents is that the minimum DNBR remains above the design limit. The DNBR design limit for each accident is identified in the analysis results discussion. The RCP locked rotor accident is categorized as a limiting fault. The acceptance criteria for the RCP locked rotor accident are that any fuel damage calculated to occur must be of a sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability, that the peak RCS pressure does not exceed 110% (2750 psig) of the design pressure, and that the calculated offsite doses are less than 100% of the 10CFR Part 100 limits. To evaluate the third criterion on offsite doses, the extent of fuel failures are quantified with the assumption that any fuel pin that exceeds the DNBR limit is considered failed. The fuel failure results are then used in the offsite dose calculations to verify that the offsite dose criteria are satisfied. The results of the locked rotor analysis demonstrates that the peak RCS pressure limit is not challenged.

15.6.2 Four RCP Coastdown from Four RCP Initial Conditions Analysis

The RETRAN system thermal-hydraulic analysis results are valid for the full core with Mk-B-HTP fuel.

The 4/4 RCP coastdown accident analysis results are shown in Figure 15-19, Figure 15-20, Figure 15-21, Figure 15-22, Figure 15-23, and Figure 15-24, and the sequence of events is given in Table 15-40. The Mk-B-HTP fuel type is analyzed. Since the transient minimum DNBR occurs near the time of reactor trip, the duration of the analysis is 20 seconds. The flow in both loops (Figure 15-19) behaves identically since the 4/4 RCP coastdown event is essentially symmetrical. The loop flows decrease towards zero flow during the transient. The pump monitor trip function trips the reactor at 0.9 seconds. The core thermal power (Figure 15-20) follows the trend of the neutron power with a thermal delay. The hot and cold leg temperatures (Figure 15-21) change only slightly in response to the change in flow during the transient. The pressurizer level (Figure 15-22) increases due to the increase in the RCS average temperature, and then decreases following the reactor trip. RCS pressure (Figure 15-23) increases initially due to the increase in pressurizer level, and decreases post-trip. The transient minimum DNBR (Figure 15-24) of 1.818 occurs at 2.1 seconds for a full core with Mk-B-HTP fuel. The minimum DNBR value is above the design limit.

15.6.3 Two RCP Coastdown from Four RCP Initial Conditions Analysis

The RETRAN system thermal-hydraulic analysis results are valid for the full core with Mk-B-HTP fuel.

The results of the 4/2 RCP coastdown accident analysis with the tripped RCPs in the same loop are presented since it is the bounding event for the four RCP initial conditions. The results are shown in Figure 15-25 and Figure 15-119, Figure 15-120, Figure 15-121, Figure 15-122, Figure 15-123, and the sequence of events is given in Table 15-41. The Mk-B-HTP fuel type is

analyzed. Since the transient minimum DNBR occurs near the time of reactor trip, the duration of the analysis is 20 seconds. The transient behavior of many of the key parameters trend those of the 4/4 RCP coastdown accident. The flux/flow imbalance trip function trips the reactor at 4.2 seconds. The core flow (Figure 15-25) decreases after the RCPs trip, and approaches the equilibrium two RCP flowrate at the end of the analysis. The faulted loop flow decreases toward zero flow, while the intact loop flow increases from its initial value. The hot leg temperatures (Figure 15-120) change only slightly in response to the change in flow during the transient. The cold leg temperatures in the affected loop decrease due to the decrease in primary flow, and then increase due to the post- trip increase in steam pressure. The cold leg temperatures in the unaffected loop initially remain stable and then increase due to the flow reversal in the loop. The transient minimum DNBR (Figure 15-123) of 1.68 occurs at 4.9 seconds for a full core with Mk- B-HTP fuel. The minimum DNBR value is above the design limit.

15.6.4 One RCP Coastdown from Three RCP Initial Conditions Analysis

The RETRAN system thermal-hydraulic analysis results are valid for the full core with Mk-B-HTP fuel.

The results of the 3/1 RCP coastdown accident analysis with the tripped RCP in the same loop as the initially idle RCP are presented since it is the bounding event for the three pump initial conditions. The results are shown in Figure 15-124, Figure 15-125, Figure 15-126, Figure 15-127, Figure 15-128, and Figure 15-129 and the sequence of events is given in Table 15-42. Since the transient minimum DNBR occurs near the time of reactor trip, the duration of the analysis is 20 seconds. The transient behavior of many of the key parameters trend those of the 4/2 RCP coastdown accident. The flux/flow imbalance trip function trips the reactor at 5.0 seconds. The RCS flow transient (Figure 15-124) approaches the two RCP equilibrium flowrate at the end of the analysis. While the affected loop flow decreases and reverses direction, the intact loop flow increases from its initial value. The transient minimum DNBR (Figure 15-129) of 1.97 occurs at 5.5 seconds for a full core with Mk-B-HTP fuel. The minimum DNBR value is above the design limit.

15.6.5 Locked Rotor from Four RCP Initial Conditions Analysis

The RETRAN system thermal-hydraulic analysis results are valid for the full core with Mk-B-HTP fuel.

The locked rotor accident from four RCP initial conditions analysis results are shown in Figure 15-130, Figure 15-131, Figure 15-132, Figure 15-133, Figure 15-134, and Figure 15-135, and the sequence of events is given in Table 15-43. Mk-B-HTP fuel type is analyzed. Since the transient minimum DNBR occurs near the time of reactor trip, the analysis is terminated at 10 seconds. The core flow (Figure 15-130) rapidly decreases after the locked rotor occurs, and approaches the equilibrium three RCP flowrate at the end of the analysis. The locked rotor cold leg flow rapidly decreases to a negative value, and the other cold leg flow increases towards the three RCP flowrate. The flux/flow trip function trips the reactor at 1.7 seconds. The core thermal power (Figure 15-131) follows the trend of the neutron power with a thermal delay. The hot leg temperatures (Figure 15-132) increase initially due to the decrease in flow. After the reactor trips, the hot leg temperatures begin to decrease. The cold leg temperature in the affected loop decreases slightly due to the decrease in primary flow. The cold leg temperature of the unaffected loop remains stable initially, and then increases post-trip due to the increase in steam pressure. The pressurizer level (Figure 15-133) increases initially due to the increase in RCS temperatures, and then decreases post-trip. The RCS pressure response (Figure 15-134)

trends with the change in pressurizer level. The limiting transient minimum DNBR (Figure 15-135) of 1.41, which occurs at 2.2 seconds, for the Mk-B-HTP fuel is equal to the design limit. A fuel pin census analysis is performed to determine if DNBR margin exists or the number of fuel pins that exceed the DNBR limit. A range of pin radial peaks and axial shapes are assumed to determine the peaking factors at which the DNBR limit is exceeded. These limiting peaking factors are the maximum allowable radial peak (MARP) limits. Each fuel pin in the core is then evaluated against the MARP limits at the limiting DNBR statepoint to determine if the DNBR limit is exceeded. All fuel pins that exceed the DNBR limit are assumed to experience cladding failure and are counted in the source term for the offsite dose calculation. The results of the fuel pin census analysis for the locked rotor accident from four RCP initial conditions is that DNBR margin exists for all of the fuel pins. Due to no fuel failures, the offsite dose consequences for the locked rotor accident are bounded by the offsite dose consequences for the steam line break accident.

The peak maximum RCS pressure is 2501 psig, which is well below 110% of the design pressure (2750 psig).

15.6.6 Locked Rotor from Three RCP Initial Conditions Analysis

The system thermal-hydraulic analyses were performed for a full core loaded with Mk-B-HTP fuel.

The locked rotor accident from three RCP initial conditions (Table 15-34) analysis results are shown in Figure 15-136, Figure 15-137, Figure 15-138, Figure 15-139, Figure 15-140, and Figure 15-141, and the sequence of events is given in Table 15-44. Since the transient minimum DNBR occurs near the time of reactor trip, the analysis is terminated at 10 seconds. The analysis results are similar to those of the four RCP initial condition analysis. The flows in the unaffected loop and the core (Figure 15-136) approach the two RCP equilibrium flowrates at the end of the analysis. The transient minimum DNBR (Figure 15-141) of 1.446 occurs at 2.4 seconds for the full core with Mk-B-HTP fuel. Both minimum DNBR values are above the design limits. A fuel pin census analysis is performed. The results of the fuel pin census analysis for the locked rotor accident from three RCP initial conditions is that DNBR margin exists for all of the fuel pins. Due to no fuel failures, the offsite dose consequences for the locked rotor accident are bounded by the offsite dose consequences for the steam line break accident.

15.6.7 Natural Circulation Capability Analysis

The natural circulation capability analysis determines the stable natural circulation flowrates for a range of post-trip decay heat values. The natural circulation flowrates are shown to be greater than the decay heat power levels on a percentage basis, thereby limiting the temperature rise across the core to less than that at full power conditions. Therefore, adequate core cooling will be maintained during natural circulation.

Decay Heat Power (MW_{th})	(% Power)	Natural Circulation Flowrate (% Full Flow)
80	3.1	3.8
70	2.7	3.6
60	2.3	3.5
50	1.9	3.3

Decay Heat Power (MW_{th})	(% Power)	Natural Circulation Flowrate (% Full Flow)
40	1.6	3.0
30	1.2	2.7
20	0.8	2.4
10	0.4	1.9

15.6.8 Environmental Consequences

The radiological consequences of a locked rotor accident are bounded by the consequences of the large main steam line break accident.

15.6.9 Conclusions

The results of the RCP coastdown accident analyses show that the limiting RCP coastdown event is two RCPs coasting down from a four RCP initial condition. The minimum DNBR is 1.68 for a full core with Mk-B-HTP fuel. The minimum DNBR is above the design limit. The results of the locked rotor accident analyses show that the limiting locked rotor event is from a four RCP initial condition. The results of a pin census analysis for the locked rotor show that DNBR margin exists for all of the fuel rods. Therefore, no fuel rod failures are assumed in the offsite dose analysis. The results of the locked rotor analysis demonstrate that the peak RCS pressure limit is not challenged. The peak maximum RCS pressure is 2501 psig, which is far below the design pressure. The results of the natural circulation capability analysis show adequate flow for core cooling and decay heat removal by natural circulation after all RCPs trip. All of the acceptance criteria are met.

15.6.10 References

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15.7 Control Rod Misalignment Accidents

15.7.1 Identification of Causes and Description

Control rods are normally grouped into patterns which maintain a symmetric core power distribution. A mechanical or electrical failure can cause a control rod to become misaligned from its group, causing an asymmetric reactivity distribution and, if the control rod is stuck, a reduction in the total available control rod worth for shutdown of the reactor. Three modes of misalignment can occur. The first mode, the statically misaligned rod accident, occurs during withdrawal or insertion of a control rod group when one rod becomes stuck at some position as the rod group continues in motion. This condition will affect the power distribution in the core and could lead to excessive power peaking. The second mode of misalignment, the stuck rod accident, can occur on reactor trip if one rod fails to insert. This condition requires an evaluation to determine that sufficient negative reactivity is available for tripping the reactor when considering the maximum worth stuck rod. The third mode, the dropped rod accident, can occur when one rod drops partially or fully into the core. The resulting plant transient response is a rapid reduction in power and a possible subsequent increase in power due to a negative moderator coefficient of reactivity. The expected plant response is that the Integrated Control System (ICS) will respond to an indicated dropped control rod by initiating a power runback and by inhibiting control rod withdrawal. A reactor trip may occur on variable low pressure-temperature for some dropped rod accidents.

For the statically misaligned rod accident, the core designs are evaluated to confirm that the resulting core power distribution is acceptable. For the stuck rod accident, each core design is required to be capable of maintaining a 1% $\Delta k/k$ shutdown margin at hot shutdown conditions with the assumption of the maximum worth rod stuck in the fully withdrawn position. The dropped rod accident is analyzed for a set of dropped rod worths for initial conditions of 102% of 2568 MWth with four reactor coolant pumps (RCPs) in operation, and for 75% of 2568 MWth with three RCPs in operation. Physics parameters for the beginning-of-cycle (BOC) condition are analyzed. The expected action taken by the ICS on indication of a dropped rod is to inhibit control rod withdrawal and to run back power demand to 55 percent of rated load at 1 percent per minute. This non-safety action by the ICS is not credited in the analysis. The ICS is assumed to respond to the decrease in reactor power by withdrawing control rods to meet the load demand, which is a conservative assumption. A reactor trip on high flux or flux/flow/imbalance may occur for some cases. The system analysis determines the transient core boundary conditions for the detailed core thermal-hydraulic analysis. The results presented model the replacement steam generators. The analysis methodology and the computer codes used in this analysis are given in Table 15-33. The initial conditions are given in Table 15-34. The Reactor Protective System (RPS) and Engineered Safeguards Protective System setpoints and delay times are given in Table 15-35.

Due to the asymmetric core power distribution resulting from the dropped rod, the excore power range flux channels which input to the RPS high flux trip function will indicate different transient power responses. The limiting single failure for the dropped rod analysis is the excore power range flux channel adjacent to the quadrant with the highest indicated core power level. This assumption results in the third highest (or second minimum) excore flux channel determining whether the high flux trip setpoint is reached based on the 2/4 RPS logic design.

The three identified modes of control rod misalignment accidents are considered to be faults of moderate frequency. The acceptance criteria for these accidents are that the minimum DNBR remains above the design limit, that the centerline fuel melt limit is not exceeded, and that the peak RCS pressure does not exceed 110% (2750 psig) of design pressure. Since this event

results in a minor RCS pressurization which does not approach the limit, only the minimum DNBR and centerline fuel melt acceptance criteria are of concern.

15.7.2 Dropped Rod Analysis

The limiting dropped rod accident is a 20 pcm dropped rod from full power at BOC conditions. The RETRAN system thermal-hydraulic analysis results are valid for a full core with Mk-B-HTP fuel. The duration of the analysis is less than 40 seconds (see Table 15-45), which is sufficient for the time of minimum DNBR. The transient response is shown in Figure 15-26, Figure 15-27, Figure 15-28, Figure 15-143 and Figure 15-144, and the sequence of events is given in Table 15-45. The initial decrease in reactor power (Figure 15-26) is caused by the reactivity inserted by the dropped rod. The ICS response, due to the asymmetric power distribution, causes control rods to be withdrawn and results in an increase in reactor power. Hot and cold leg temperatures (Figure 15-27) increase at a steady rate due to the power mismatch between reactor power and steam generator heat removal. The trends of pressurizer level (Figure 15-28) and RCS pressure (Figure 15-143) reflect this power mismatch. The maximum RCS pressure is less than 2350 psig. The transient minimum DNBR (Figure 15-144) of 1.878 occurs at 77.7 seconds for a full core of Mk-B-HTP fuel. This minimum DNBR value is greater than the design limit.

15.7.3 Statically Misaligned Rod Analysis

The results of the generic evaluation of the statically misaligned rod event show that this event is bounded by the dropped rod event.

15.7.4 Conclusions

The stuck rod accident cannot result in insufficient negative reactivity insertion on reactor trip due to the core design criteria. The statically misaligned rod accident has been shown to be bounded by the dropped rod accident. The minimum DNBR is shown above and greater than the design limit. No fuel centerline melt is predicted. The RCS pressure transient does not approach the peak primary pressure limit. All of the acceptance criteria are met.

15.7.5 References

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15.8 Turbine Trip Accident

15.8.1 Identification of Causes and Description

The turbine trip accident is caused by events including a generator trip, low condenser vacuum, loss of turbine lubrication oil, turbine overspeed, main feedwater pump trip, high steam generator level, or a reactor trip. The rapid closure of the main turbine stop valves results in a rapid increase in the secondary pressure and temperature. This degradation in the secondary heat sink creates a mismatch between power generated in the Reactor Coolant System (RCS) and heat removed by the secondary. As a result, the RCS temperature and pressure increase. The expected plant response to a turbine trip would be an immediate reactor trip initiated by the turbine trip signal. The Turbine Bypass System (TBS) and main steam code safety valves would then relieve steam in order to control the post-trip steam generator pressures. RCS pressure would be controlled by the pressurizer spray, PORV, and heaters. In addition, feedwater would be automatically controlled by the Integrated Control System (ICS) to maintain the post-trip steam generator levels at setpoint.

The turbine trip accident is analyzed from a full power initial condition at beginning-of-cycle. The analysis assumes that the pressurizer spray, pressurizer PORV, and the TBS are inoperable. In addition, the pressurizer and main steam code safety valves are modeled using conservative assumptions for drift, blowdown and valve capacity that minimize relief flow. The anticipatory reactor trip on turbine trip is not credited. Main feedwater is isolated coincident with the turbine trip in order to maximize the steam generator pressure. Also, no credit is taken for the Emergency Feedwater System (EFW), since the peak pressure will be reached before EFW flow can start and have an effect on the transient response. The results presented model the replacement steam generators. The analysis methodology and the computer codes used in this analysis are given in Table 15-33. The initial conditions are given in Table 15-34. The Reactor Protective System and Engineered Safeguards System setpoints and delay times are given in Table 15-35.

No single failure has been identified which adversely impacts the results of the turbine trip analysis.

The turbine trip accident is considered to be a fault of moderate frequency. The acceptance criteria for this accident are that the minimum DNBR remains above the design limit, and that the peak RCS pressure does not exceed 110% (2750 psig) of design pressure. The DNBR limit is not challenged since the increase in RCS pressure more than offsets the slight increase in RCS temperature.

15.8.2 Analysis

The turbine trip accident analysis results are shown in Figure 15-145, Figure 15-146, Figure 15-147, Figure 15-148, and Figure 15-149, and the sequence of events is given in Table 15-46. Mk-B-HTP fuel type is analyzed. The analysis duration of 40 seconds is sufficient to demonstrate the peak RCS pressure. The closure of the main turbine stop valves results in a rapid increase in steam line pressure (Figure 15-145) and temperature. The RCS hot and cold leg temperatures (Figure 15-146) increase due to the increasing secondary side temperature. The increase in RCS temperatures causes pressurizer level (Figure 15-147) and RCS pressure (Figure 15-148) to increase, resulting in a reactor trip on high RCS pressure at 3.6 seconds. Following the reactor trip, the RCS temperatures, pressurizer level, and RCS pressure all decrease towards the post-trip values. Figure 15-149 shows the power remains constant prior to

trip. The RCS pressure at the bottom of the reactor vessel reaches a maximum value that is below 2750 psig.

15.8.3 Conclusions

The turbine trip accident analysis results in a peak RCS pressure that is below 2750 psig. All of the acceptance criteria are met.

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15.9 Steam Generator Tube Rupture Accident

15.9.1 Identification of Causes and Description

The steam generator tube rupture (SGTR) accident is caused by a double-ended rupture of a single steam generator tube. The expected plant response is as follows. The tube rupture initiates a blowdown of primary coolant into a steam generator. The plant response to this event is similar to a small break LOCA in that the Reactor Coolant System (RCS) pressure and pressurizer level would decrease as coolant inventory is lost through the ruptured steam generator tube. Makeup flow to the RCS would increase in response to the decrease in pressurizer level. The Integrated Control System (ICS) would reduce main feedwater (MFW) to the ruptured steam generator to compensate for the break flow. Without operator action, the reactor would trip on the variable low pressure-temperature trip function. With operator action, actions would be taken to initiate a rapid shutdown of the reactor. This would be accomplished by making up for the loss of RCS inventory through the break with flow from the High Pressure Injection System (HPIS). When the reactor power level has been reduced to below the capacity of the Turbine Bypass System (TBS), a manual reactor trip would be performed. Following the reactor trip, the TBS would relieve steam to control steam generator pressure. MFW would be automatically controlled by the ICS to maintain the post-trip steam generator level at setpoint. The operator would then isolate the ruptured steam generator and depressurize the RCS to decrease the subcooled margin, thereby minimizing primary-to- secondary leakage. A plant cooldown and depressurization would then be initiated using the TBS and the unaffected steam generator to bring the plant to the conditions where the Low Pressure Injection System (LPIS) can be aligned for decay heat removal, and break flow could then be terminated. The ruptured steam generator would be steamed and/or drained as necessary to prevent overfill during the course of the event.

The SGTR accident is analyzed from a full power initial condition at end-of-cycle with maximum decay heat. Analysis assumptions are selected to maximize the environmental consequences. Offsite power remains available. A conservatively long delay time is assumed for the Reactor Protective System to trip the reactor to maximize the pre-trip primary coolant leakage into the ruptured steam generator. It is further assumed that the operator takes action to maintain RCS pressure and pressurizer level at the initial conditions such that the primary-to-secondary leakage is maximized. The reactor is then assumed to trip from a full power condition which results in the largest post-trip steam release through the main steam safety valves (MSSVs). The MFW pumps are assumed to trip on reactor trip to minimize the secondary heat sink, which actuates the emergency feedwater (EFW) pumps. A penalty for the turbine-driven EFW pump is taken in the analysis since the steam supply to its turbine originates from the SG with the tube rupture and exhausts directly to the atmosphere. However, no EFW flow from the turbine-driven pump is credited. The non-safety TBS is also not credited in the analysis. The results presented model the replacement steam generators. The analysis methodology and the computer codes used in this analysis are given in Table 15-33. The initial conditions are given in Table 15-34. The RPS and Engineered Safeguards Protective System setpoints and delay times are given in Table 15-35.

The analysis credits the non-safety manual steam line atmospheric dump valves (ADVs) to cool down the plant. The single failure assumed in this event is the EFW control valve on the unaffected steam generator failing to open following the reactor trip. This results in only the ruptured steam generator being available for cooling down the plant until operator action is taken to establish an alternate EFW alignment. The following operator actions are credited during this event:

1. Immediate action to maximize HPI flow.
2. Identify the failed-closed position of the EFW control valve and restore EFW to the unaffected steam generator. A delay time of 23 minutes after reactor trip is assumed.
3. The ruptured steam generator is identified 10 minutes after EFW restoration to the unaffected steam generator.
4. Cooldown of the plant to 532°F begins 52 minutes after the ruptured steam generator is identified.
5. The ruptured steam generator is isolated after the plant has been cooled down to 532°F.
6. The RCS subcooled margin is minimized 12 minutes after the ruptured steam generator is identified.
7. One reactor coolant pump (RCP) in the loop without the pressurizer is tripped off 20 minutes after the RCS has been cooled down to 532°F. Operators trip one RCP in loop with pressurizer at 400°F.
8. A shift changeover delay of one hour is assumed after the RCS has been cooled down to 532°F and one RCP in the loop without the pressurizer has been tripped.
9. An RCS cooldown to 450°F begins after the shift changeover is complete.
10. Cooldown of the RCS is stopped upon reaching 450°F while the RCS boron concentration is verified. A delay time of 90 minutes is assumed.
11. Boration of the RCS is performed to achieve the cold shutdown boron concentration requirement. A delay time of 30 minutes is assumed.
12. Cooldown to decay heat removal conditions resumes 5 minutes after the cold shutdown boron concentration has been achieved.
13. Periodic steaming of the ruptured steam generator is performed to prevent water from entering the steam lines.
14. A 90 minute delay is assumed to align the LPIS for decay heat removal. RCS temperature and pressure are held constant during this time.

The steam generator tube rupture accident is considered to be a limiting fault event. The acceptance criterion for this event is that the calculated doses at the site boundary are less than 100% of the 10CFR100 guidelines.

15.9.2 Analysis

The SGTR accident analysis results are shown in Figure 15-150, Figure 15-151, Figure 15-152, Figure 15-153, Figure 15-154, Figure 15-155, and Figure 15-156, and the sequence of events is given in Table 15-47. The duration of the analysis is until the plant has been cooled down and steam releases to the atmosphere have terminated, which is 40,725 seconds (11.3 hours). As a result of the tube rupture and immediate operator action to increase HPIS flow to compensate for the loss of RCS inventory, RCS conditions remain relatively stable until the RPS is assumed to trip the reactor at 1200 seconds. The reactor power response is shown in Figure 15-150. MFW flow is automatically throttled to compensate for the break flow (Figure 15-151) entering the ruptured steam generator. A normal post-trip response occurs, with RCS pressure (Figure 15-152) and pressurizer level (Figure 15-153) decreasing due to RCS shrinkage and steam generator pressures (Figure 15-154) increasing to the MSSV lift setpoints. MFW flow is lost on reactor trip. Steam generator levels (Figure 15-155) decrease to the post-trip setpoints, and

then the unaffected steam generator continues to boil down to a dried out condition due to the failure of its EFW control valve to open. Post-trip heat removal is provided by the ruptured steam generator until an alternate EFW flowpath to the unaffected steam generator is aligned at 2580 seconds. After restoration of EFW to both steam generators, the ruptured steam generator is identified at 3180 seconds due to the EFW flow imbalance between the steam generators. The RCS subcooled margin is reduced at 3900 seconds to minimize primary-to-secondary leakage. This is conservatively assumed to be accomplished using pressurizer spray which is slower than other potentially available means of depressurizing the RCS (i.e.; RCS PORV, Auxiliary Spray). At 6300 seconds, the unit is cooled down to 532°F (Figure 15-156) using the ADVs on both steam lines. The ruptured steam generator and EFW to the ruptured steam generator are isolated after reaching 532°F (~7,040 seconds), with all steam release flowpaths being isolated by 8,240 seconds. After one RCP is tripped in the loop without the pressurizer, the RCS is held at a constant temperature and pressure while a shift changeover occurs. During the shift changeover, steaming of the ruptured steam generator begins due to the water level reaching the high level setpoint (10,621 seconds). Steaming the ruptured steam generator continues for the remainder of the analysis. The plant cooldown is resumed following the shift changeover, with RCS temperatures reaching 450°F at 15,191 seconds. RCS Boron concentration determination is initiated and boration to cold shutdown conditions is accomplished by 22,391 seconds, with the plant cooldown resuming at 22,691 seconds. LPIS decay heat removal conditions are reached at 31,455 seconds, where RCS pressure and temperature are held constant while this system is aligned. The plant cooldown continues at 36,855 seconds, with the RCS reaching 215°F at 40,725 seconds. The analysis is terminated at this time since steam releases to the atmosphere have stopped.

15.9.3 Environmental Consequences for the Steam Generator Tube Rupture

The postulated accidents involving release of steam from the secondary system do not result in a significant release of radioactivity unless there is leakage from the RCS to the secondary system in the steam generators as with the SGTR. A conservative analysis of the potential offsite doses resulting from a SGTR accident is presented assuming a pre-existing primary to secondary leakage. This activity is released to the environment by releases associated with the normal operation of plant equipment or the operation of plant equipment as intended in response to the accident, and as part of the subsequent cooldown activities.

Two RCS source terms are examined as part of this analysis. The first models an initial RCS activity of one percent of the core averaged isotopic inventory. This source term bounds the allowed normal RCS DEI activity concentration permitted by Technical Specifications. The second source term models the maximum DEI activity concentration permitted by Technical Specifications for an iodine spike at full power. This “pre-existing” spike is postulated to occur at the time of accident initiation. Both of these source terms are modeled to be released instantaneously and homogeneously such that the RCS activity is in equilibrium at the start of the accident. Both source terms also bound Technical Specification limits for non-iodine isotopes. Source term isotopics are based upon fuel depletion and projected fission product inventories at the end of the cycle with the maximum thermal power uncertainty applied.

An initial source term is also modeled for the secondary side. The maximum Technical Specification allowed DEI concentration is modeled to be present in the secondary side water, the steam generators and any makeup water supplied to the unit. Thus, the secondary side is essentially modeled as an infinite source of water at the secondary side Technical Specification DEI concentration limit.

In order to transport and release primary activity to the environment, a primary to secondary release path is modeled in the steam generators. This path is postulated to exist at the start of

the accident, but is not caused by the SGTR. The tube leakage into the unaffected steam generator modeled bounds the maximum allowed tube leakage rate into one steam generator. The affected steam generator is modeled with a break flow that is based on the thermal/hydraulic model.

The thermal/hydraulic model discussed in the previous sections is used as the basis for the plant response and steam releases modeled in the environmental analysis. The plant is initially operating in a normal mode at full power (plus maximum thermal power uncertainty) with primary to secondary leakage. When the break initiates, the activities in the primary and secondary side are modeled to be instantaneously and homogeneously released to their respective systems. Shortly after the break initiates, the reactor is automatically tripped and radioactive decay (and daughter product production) is begun in the model. The steam generators begin to discharge their activity directly to the environment through the Atmospheric Dump Valves (ADVs).

In order to maximize releases to the environment, the condenser is assumed to not be available. This requires that the unit be cooled down using the steam generators by discharging steam from the steam generators directly to the environment through the ADVs. No credit is taken for the condenser and no partitioning credit is taken for releases.

The steam generator tube rupture causes the Turbine Driven Emergency Feedwater Pump (TDEFWP) to start and briefly supply makeup water. The TDEFWP is driven by steam from the Main Steam System or the Auxiliary Steam System and exhausts directly to the environment, and therefore, is a release path that is included in the environmental analysis.

Since Oconee Nuclear Station is a B&W designed plant, it uses once through steam generators which provide for vertical tubing which carries primary coolant from the top of the generator to its bottom while exchanging heat with the secondary fluid on the shell side. Because of this tubing arrangement, the tube leakage is modeled to occur above the secondary water mass in the steam generator. Therefore, no credit is taken for iodine partitioning in the steam generator. No credit is taken for iodine plateout in the steam lines or any other surface.

When the thermodynamic conditions are met for the Low Pressure Injection (LPI) system to remove decay heat from the primary, cooldown releases from the ADVs cease and decay heat removal is accomplished by the LPI system. Primary to secondary leakage and its release to the atmosphere continue until the temperature of the primary water leaking is less than the boiling point for water at atmospheric conditions. At this point all releases of activity from the plant model cease.

Offsite atmospheric dispersion factors from the Updated Final Safety Analysis Report Chapter 2 were used. Dose conversion factors from Federal Guidance Reports 11 and 12 were used.

Based upon this model, releases of activity to the environment from the primary and secondary systems can be calculated and used to calculate doses offsite at the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ). The doses calculated meet the regulatory criteria of 10 CFR 100 for each of the source terms examined. The results are presented in Table 15-16.

15.9.4 Conclusions

The steam generator tube rupture accident is analyzed to provide conservative inputs to the environmental consequences analysis. The results of the environmental consequences analyses are within the 10CFR100 limits. All of the acceptance criteria are met.

15.9.5 References

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15.10 Waste Gas Tank Rupture Accident

15.10.1 Identification of Accident

Rupture of a waste gas tank would result in the release of the radioactive contents of the tank to the plant auxiliary building ventilation system and to the atmosphere through the unit vent. The release is assumed to occur over a two hour period to maximize the exclusion area boundary dose. Dose to a receptor at the site boundary and the control room dose evaluated.

15.10.2 Analysis and Results

A tank is assumed to contain the maximum inventory expected based on a technical specification limit which requires that offsite dose from a tank rupture be limited to 500 millirem. The tank inventory assumed in this analysis is far greater than the expected operational inventory and is not based on actual operation of the system. The shared unit 1 & 2 tank is considered as the limiting case and is assumed to contain the following noble gas inventory.

Isotope	Waste Gas Tank Inventory Activity (Ci)
Kr-85m	888
Kr-85	68,657
Kr-87	484
Kr-88	1,519
Xe-133m	2,560
Xe-133	186,345
Xe-135m	282
Xe-135	5,344

The Total Effective Dose Equilivant from a puff release of this inventory to the site boundary is calculated to be 0.44 Rem at the exclusionary boundary and 0.048 Rem at the Low Population Zone boundary. Control Room Dose is less than 0.338 Rem TEDE.

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15.11 Fuel Handling Accidents

15.11.1 Identification of Accident

Spent fuel assemblies are handled entirely under water. The Core Operating Limits Report, refueling boron concentration, ensures shutdown margin is maintained. Procedures ensure that fuel assemblies are in configurations such that this shutdown margin is maintained In the spent fuel storage pool, the fuel assemblies are stored under water in storage racks with a minimum boron concentration as specified by the Core Operating Limits Report (COLR) in the pool water. Under these conditions, a criticality accident during refueling is not considered credible. Fuel handling consists of all fuel assembly shuffling and transfer operations between the reactor, the spent fuel pool, the fuel shipping casks, and dry storage transfer cask. Mechanical damage to the fuel assemblies during transfer operations is possible but improbable. The mechanical damage type of accident is considered the maximum potential source of activity release during refueling operations.

15.11.2 Analysis and Results

15.11.2.1 Base Case Fuel Handling Accident in Spent Fuel Pool

During fuel handling operations, it is possible that a fuel assembly can be dropped, causing mechanical damage with a subsequent release of fission products. To conservatively evaluate the offsite dose consequences of such an accident, conservative assumptions are made. The following analysis assumes the accident occurs within the spent fuel pool building.

The fuel assembly gap inventory is assumed to contain a fission product inventory from a maximum burned fuel assembly at a radial peaking factor of 1.65. The gap fractions used are from Reg. Guide 1.183 and the reactor has been shutdown for 72 hours, which is the minimum time for RCS cooldown, reactor closure head removal, and removal of the first fuel assembly. For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals. The actual isotopic curie contents are listed in Table 15-1. It is also assumed that all 208 fuel pins are mechanically damaged such that the entire gap inventory is released to the surrounding water. Since the fuel pellets are cold, only the gap inventory is released. The maximum fuel rod internal pressure in the spent fuel is 1300 psig as used in the computer code TACO3 to determine the fuel rod internal pressure.

The gases released from the damaged fuel assembly pass upward through the spent fuel pool water prior to reaching the Auxiliary Building atmosphere. Noble gases are assumed to not be retained in the pool water. According to Reg Guide 1.183, an iodine decontamination factor of 200 can be used for water depths of 23 feet or greater. Since the spent fuel pool racks are at an elevation of 816.5 feet and the minimum water level in the Spent Fuel Pool is equal to or greater than 837.84 feet, there is a minimum of 21.34 feet of water over the fuel storage racks, including instrument error. An experimental test program (Reference 2) evaluated the extent of removal of iodine released from a damaged irradiated fuel assembly. Iodine removal from the released gas takes place as the gas rises through the water. The extent of iodine removal is determined by mass transfer from the gas phase to the surrounding liquid and is controlled by the bubble diameter and contact time of the bubble with the water. The following analytical expression is given as a result of this experimental test program:

Iodine Decontamination Factor (DF) = $73\ e^{0.313\ (t/d)}$

Where:

t = bubble rise time, seconds

d = effective bubble diameter, cm

Since the minimum water depth over a dropped fuel assembly is less than 23 feet (21.34 feet), the assumed iodine DF must be less than 200, according to Reg. Guide 1.183, and calculated with comparable conservatism. Using the above relationship, with a water depth of 21.34 feet, a comparable DF is equal to 183 (Revision 1).

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The activity released from the water's surface is released within a two-hour period as a ground release. The atmospheric dilution is calculated using the two-hour ground release dispersion factor of 2.2 x 10^4 sec/m^3.

The total effective dose equivalent (TEDE) doses are given in Table 15-16. These values are below the limits given in Regulatory Guide 1.183.

15.11.2.2 Base Case Fuel Handling Accident Inside Containment

The offsite dose consequences for a fuel handling accident inside containment were evaulated per the guidance given in Reg. Guide 1.183. Since the shallow end of the fuel transfer canal is at an elevation of 816.5 feet, the same iodine decontamination factor used for the Fuel Handling Accident in the Spent Fuel Pool is used for the Fuel Handling Accident inside Containment. The activity released from the refueling water is released as a ground release, which has an atmospheric dispersion factor of 2.2 x 10^4 sec/m^3. There is no credit taken for any containment closure/integrity resulting in the released activity from the refueling water going straight outside.

Using the fuel assembly gap inventory in Table 15-1, and assuming all 208 fuel pins are damaged, the calculated doses are appropriately within the guidelines given in Regulatory Guide 1.183. For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals. The limiting doses for a fuel handling accident for a single fuel assembly event are given in Table 15-16.

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15.11.2.4 Shipping Cask Drop Accidents

Fuel shipping casks are used to transport irradiated fuel assemblies from the site and also between the Oconee 1 and 2 spent fuel pool and the Oconee 3 spent fuel pool.

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The worst case fuel handling accident sequence in which the fuel shipping cask impacts on the irradiated fuel assemblies in a spent fuel pool is evaluated. At no time is the cask suspended above the spent fuel; however, it is credible that with failure of the cask hoist cable that the cask, yoke, hook, and load block could, as a result of an eccentric drop, deflect and fall into the spent fuel pool and impact on top of the assemblies in the pool. The analysis is performed separately for the shared Unit 1 and 2 spent fuel pool and the Unit 3 spent fuel pool. In the first part of the analysis, the number of fuel assemblies damaged as a result of the cask drop is found. Subsequently the radiological consequences of the damaged assemblies are determined.

The following conservative assumptions are employed for determining the number of fuel assemblies damaged.

1. The cask, lifting yoke and load block are free to fall from elevation 844 ft., the top of the spent fuel pool, to elevation 816 ft. 5 in., the top of the fuel storage racks.
2. The drag on the cask, lifting yoke and load block from falling through 25.5 ft. of water is neglected.
3. The ability of the fuel storage cells to absorb energy beyond the point of elastic buckling is neglected.
4. The energy which is expended in deformation of the rack interconnecting members is neglected.
5. A deformed fuel storage cell results in the total loss of integrity of one fuel assembly.
6. The projected areas of the cask, lifting yoke and load block are oriented to contact the maximum number of fuel assemblies.

Using the above assumptions, the falling cask, lifting yoke, and load block will have 2.093×10^6 ft-lbf of kinetic energy at the instant of impact with the storage racks. This energy must be absorbed by the strain energy in the storage racks. For additional conservatism it is assumed that the storage racks which are directly impacted by the falling load in turn buckle and deflect into adjacent racks until the total energy of the falling cask is absorbed. The Unit 1 and 2 spent fuel pool contains 154 fuel storage positions under the direct impact area, with a total of 576 spent fuel assemblies which can potentially suffer a loss of integrity during a cask drop accident. The Unit 3 pool contains 156 fuel storage positions under the projected impact area, with a total of 518 assemblies which can be damaged during the accident. These analyses are based on the TN8 three element shipping cask.

Once the number of fuel assemblies which could be damaged is determined, dose analyses are performed which are consistent with Regulatory Guide 1.183, and NUREG0612. The following assumptions apply:

1. Spent fuel stored in the first 36 rows of the Unit 1 and 2 spent fuel pool closest to the spent fuel cask handling area has decayed at least 55 days. This is consistent with Technical Specification 3.7.15.a, "Plant Systems".
2. All fuel assemblies assumed damaged in excess of two full cores (354 assemblies) in the Unit 1 and 2 spent fuel pool are assumed to have decayed at least one year.
3. Spent fuel stored in the first 33 rows of the Unit 3 spent fuel pool closest to the spent fuel cask handling area has decayed at least 70 days. This is consistent with Technical Specification 3.7.15.b., "Plant Systems".
4. All fuel assemblies assumed damaged in excess of one full core (177 assemblies) in the Unit 3 spent fuel pool are assumed to have decayed at least one year.
5. The affected assemblies have the maximum core activity corresponding to a radial peaking factor of 1.2.
6. All rods of the affected assemblies are ruptured.
7. The iodine decontamination factor in pool water is 183.
8. There is no removal of activity by the spent fuel pool ventilation system filters prior to release to the environment.
9. Activity is released at ground level with an assumed χ/Q factor of 2.2×10^4 sec/m^3.

10. The fractions of noble gases and iodine in the gaps are shown below. For fuel pins which exceed the rod power/burnup criteria of Footnote 11 in RG 1.183, the gap fractions from RG 1.183 are increased by a factor of 3 for Kr-85, Xe-133, Cs-134 and Cs-137, and increased by a factor of 2 for I-131, and other noble gases, halogens and alkali metals (Reference 1).

Kr85, I131	10%, 8%
All other noble gases	5%
All other iodines	5%

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The offsite radiological consequences of the postulated cask drop accident in either spent fuel pool is within the Regulatory Guide 1.183 limits. The limiting doses for a fuel cask handling accident for a multiple fuel assembly event are given in Table 15-16.

15.11.2.5 Dry Storage Transfer Cask Drop Accident in Spent Fuel Pool Building

Dry storage transfer operations from the spent fuel pool (SFP) buildings to the Independent Spent Fuel Storage Facility (ISFSI) are routinely performed at Oconee. The major steps in the process involve transporting the transfer cask/dry storage canister (DSC) into the fuel building, placing into the SFP, loading with 24 qualified fuel assemblies, drying/sealing, and removing to the ISFSI. The potential exists for dropping the cask in the SFP area during transfer operations.

15.11.2.5.1 Criticality Analyses for Dry Storage Transfer Cask Drop Scenarios

While the transfer cask is never carried directly over spent fuel, the potential always exists for failure of the overhead crane or handling equipment. Thus, an analysis was performed assuming the cask, yoke, and yoke block are deflected into the Unit 1&2 SFP. In such a case, it was postulated that 1024 spent fuel assemblies (SFAs) would be damaged (the first 64 rows, each containing 16 SFAs). It was assumed that 220 fuel storage cells directly beneath the falling parts buckle and deflect into adjacent cells until all the energy of the dropping cask is absorbed. For a cask drop in the smaller Unit 3 SFP, it was assumed all 825 fuel cell locations would be damaged.

The potential for criticality in the SFPs was analyzed using the methodology identified in NUREG0612. It was assumed the racks and fuel were deformed such that keff was maximized. Credit was taken for pool boron and stainless steel walls to determine the keff under the assumed damage conditions. The confirmatory calculations utilized a specific neutronic analysis for each SFP with the following assumptions:

1. An infinite array of SFAs is crushed together into a geometry that optimizes keff.
2. The affected SFAs are unirradiated and have the maximum enrichment permitted for storage in the Oconee SFPs.
3. The minimum technical specification for SFP boron concentration is maintained.

The acceptance criteria for this accident per NUREG0612, is that k_{eff} will be less than or equal to 0.95 including all uncertainties. A series of calculations involving cases of varied pin pitch modeling the crushed cells and SFAs was performed. The maximum k_{eff} value determined for

the Unit 1&2 SFP was 0.9491. The maximum k_{eff} value calculated for the Unit 3 SFP was 0.9392. These analyses verify that subcriticality in the SFP will be maintained after a dry storage cask drop accident (Reference 9).

The DSC internals are designed to prevent criticality during the wet loading and unloading process. As long as the SFP boron concentration is within the limit specified in CoC 1004 for the NUHOMS Storage System and for DSCs loaded under the Site Specific License SNM2503, the DSC is drained of water within 50 hours of loading the SFAs, criticality is precluded. Strict administrative controls are in place at Oconee to ensure the SFP boron concentration is maintained above the minimum required and that the draining time for Site Specific DSC's limit is not exceeded.

The consequences of dropping the dry storage transfer cask outside the fuel building are described in the ISFSI FSAR (Reference 11, 12).

15.11.2.5.2 Potential Damage to SFP Structures from Dry Storage Transfer Cask Drop

The concrete floor slab is designed to withstand the 100 ton cask drop. However, localized concrete could be crushed and the steel liner plate punctured in the area of dry storage cask impact. For the purpose of analyzing the event, a gap of 1/64 inch for a perimeter of 308 inches in the liner plate was assumed. The calculated leakage of pool water through the gap is 21.3 gallons per day. This amount of water loss is within the capability of the SFP makeup sources.

15.11.2.5.3 Radiological Dose from Dry Storage Transfer Cask Drop

The worst radiological consequences resulting from a dry storage cask drop accident into either the Unit 1&2 or the Unit 3 SFP were analyzed. The calculation assumes a total of 1024 SFAs would be damaged in the Unit 1&2 SFP. Of this number, two full core inventories (354 SFAs) with worst case fission product concentration and less than 1 year decay time are assumed to be present. For the Unit 3 SFP, all 825 fuel cell locations are assumed to contain SFAs that would be damaged by the cask drop. One full core inventory (177 SFAs) with worst case fission product inventory and less than 1 year decay is considered to be present in the Unit 3 pool. Thus, the analysis assumes 670 and 648 SFAs, for Unit 1&2 and Unit 3 SFPs respectively, have a minimum of 1 year decay time.

Oconee Technical Specification 3.7.15.c, "Plant Systems," requires that fuel stored in the first 64 rows closest to the cask handling area be decayed a minimum of 65 days prior to movement of the dry storage transfer cask in the Unit 1&2 SFP area. Likewise, Technical Specification 3.7.15.d, "Plant Systems," requires all SFAs stored in the Unit 3 pool must be decayed a minimum of 57 days before movement of the cask is permitted in that area. The maximum fission product inventories for the iodine and noble gas nuclides of interest at times of 57 days, 65 days, and 1 year were calculated in Reference 3. This information, in conjunction with the assumed pool inventories, was used to determine the curies of each nuclide released from the postulated cask drop accidents. The total activity releases for each pool were used to determine the worst case dose consequences.

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The calculated doses are less than the Regulatory Guide 1.183 limits. Therefore, the accident dose criteria will not be exceeded for the limiting postulated dry storage cask drop accident.

15.11.3 References

1. DPC Engineering Calculation OSC7738, "Fuel Handling Accidents (FHA) Dose Analysis", dated January 28, 2010.
2. WCAP7828, "Radiological Consequences of a Fuel Handling Accident", December 1971.
3. DPC Engineering Calculation OSC9154, "Oconee Isotopic Source Term Calculations", dated July 23, 2008.
4. Deleted per 2005 Update.
5. Parker, W. O. Jr., Letter to Rusche, B. C. (NRC), November 3, 1975.
6. Parker, W. O. Jr. (Duke), Letter to Denton, H. R. (NRC), July 25, 1980.
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8. Deleted per 2005 Update.
9. DPC Engineering Calculation OSC3631 Rev 2, "Criticality Consequences of a Heavy Load Drop in the Spent Fuel Pool", dated February 7, 1996.
10. Deleted per 2002 update.
11. Oconee Nuclear Station Site Specific Independent Spent Fuel Storage Installation, Final Safety Analysis Report, Chapter 8.
12. Oconee Nuclear Station Independent Spent Fuel Storage Installation General License, Updated Final Safety Analysis Report.
13. Deleted per 2006 update.
14. Deleted per 2006 update.
15. Deleted per 2006 update.
16. Deleted per 2006 update.

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15.12 Rod Ejection Accident

15.12.1 Identification of Causes and Description

The rod ejection accident is caused by a failure of a control rod drive mechanism housing, which allows a control rod to be rapidly ejected from the reactor by the Reactor Coolant System (RCS) pressure. The control rod is ejected in 0.15 seconds from the fully inserted position. A power excursion will result, and if the reactivity worth of the ejected control rod is large enough, the reactor will become prompt critical. The resulting power excursion will be limited by the fuel temperature feedback and the accident will be terminated when the Reactor Protective System (RPS) trips the reactor on high neutron flux or high RCS pressure. RCS pressure increases due to the core power excursion, and pressurizer spray, the pressurizer PORV, and the pressurizer code safety valves will respond to mitigate the pressure increase. If a rod ejection were to occur, the nuclear design of the reactor and limits on control rod insertion will limit any potential fuel damage to acceptable levels. Cladding failure can result from the core power excursion and the highly peaked core power distribution near the ejected rod location. The failure of the control rod drive mechanism housing also constitutes a 1.50 inch diameter small-break LOCA (SBLOCA). The Emergency Core Cooling System (ECCS) will actuate on low RCS pressure or high Reactor Building pressure and will maintain core cooling. This type of SBLOCA is bounded by the limiting SBLOCA analyses presented in Sections 6.2 and 15.14.

Analyses are performed for a full core loaded with Mk-B-HTP fuel with UO_2-Gadolinium (Gad) fuel rods with different initial core conditions and number of reactor coolant pumps (RCPs) in operation. Analysis results are shown in Table 15-2. Six cases are analyzed for the full Mk-B-HTP with UO_2 Gad Fuel rods core as follows (Table 15-34). Two cases initiate at zero power (1E-7% of full power) with three RCPs in operation, at both BOC and EOC; two cases initiate at 77% power with three RCPs in operation, at both BOC and EOC; two cases initiate at 102% with four RCPs in operation, at both BOC and EOC. Since cladding failure due to exceeding the DNBR limit will result, the different possible RCP operating conditions are analyzed to bound the effect of core flowrate on DNBR. Zero power and full power are both analyzed to bound the range of ejected rod worths, initial fuel temperatures, and core power distributions. The ejected rod worth for each case is based on the power level dependent rod insertion limit including uncertainty. The negative reactivity inserted on reactor trip assumes that the most reactive control rod remains in the fully withdrawn position. The pressurizer spray and PORV are not credited for mitigating the pressure transient in the evaluation of the peak RCS pressure response. The analysis methodology and the computer codes used in this analysis are given in Table 15-33. The initial conditions are given in Table 15-34. The RPS and Engineered Safeguards Protective System setpoints and delay times are given in Table 15-35. The results presented model the replacement steam generators.

Due to the asymmetric core power distribution resulting from the rod ejection, the excore power range flux channels which input to the RPS high flux trip function will indicate different transient power responses. The analyses assume a single failure of the excore flux channel which indicates the highest power level. This assumption results in the third highest excore flux channel determining the time of reactor trip based on the 2/4 RPS trip logic design.

The rod ejection accident is considered to be a limiting fault. The acceptance criteria for the rod ejection accident analysis are that the accident will not further damage the RCS, and that the doses will be less than the 10CFR50.67 limits. The first criterion of no further damage to the RCS is interpreted to mean that the peak RCS pressure and the peak pellet radial average enthalpy both remain below a specified limit. The peak primary pressure limit is to remain within Service Limit C as defined by the ASME Code (Reference 13), which is 120% of the 2500 psig

design pressure, or 3000 psig. The peak enthalpy limit is such that the radially averaged fuel pellet enthalpy shall not exceed 280 cal/gm at any location in the core. To evaluate the second criterion of dose being within the 10CFR50.67 limits, the extent of fuel failures are quantified with the assumption that any fuel pin that exceeds the CHF DNB design limits is considered failed. The fuel failure results are used in the dose calculations to verify that the dose criteria are satisfied. The dose analysis also considers the SBLOCA release to the Reactor Building.

15.12.2 Core Kinetics Analysis

The rod ejection accident core kinetics response is determined with a three dimensional space/time analysis using SIMULATE-3K for each of the six full core Mk-B-HTP with UO_2-Gad cases. Important inputs and results for all of the cases are shown in Table 15-2. Only the ejected rod worth at BOC and EOC transients at hot zero power is large enough to achieve prompt criticality (reactivity greater than one dollar). The neutron power transients for all six cases of full core Mk-B-HTP with UO_2-Gad are shown in Figure 15-29, Figure 15-30, Figure 15-31, Figure 15-32, Figure 15-33, and Figure 15-34. For all cases the power excursion is terminated by the Doppler temperature feedback and the reactor is shut down by the reactor trip on high flux or flux/flow setpoints.

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15.12.3 Fuel Pellet Enthalpy Analysis

For each of the six rod ejection accident cases, the core power excursion and the time-dependent three-dimensional power distribution from the Mk-B-HTP with UO_2-Gad fuel rods SIMULATE-3K core kinetics analyses are used as input to the calculation of the fuel pellet peak radial average enthalpy. The results for the six cases are shown in Table 15-2. The limiting case is at 102% power and has a peak enthalpy of 134.0 cal/gm.

15.12.4 Core Cooling Capability Analysis

For each of the six rod ejection accident cases, the core power excursion from the Mk-B-HTP with UO_2-Gad fuel rods SIMULATE-3K core kinetics analysis is combined with the core flowrate, temperature, and pressure transients from the system analysis to determine the DNBR response. A range of assembly peaking factors and axial shapes are assumed to determine the peaking factors at which the DNBR limit is exceeded for each of the six cases. These limiting peaking factors are the maximum allowable radial peak (MARP) limits. Each fuel rod in the core is then evaluated against the MARP limits at the limiting DNBR statepoint to determine if the fuel rod exceeds the DNBR limit. All fuel rods that exceed the DNBR limit are assumed to experience cladding failure and are included in the source term for the offsite dose calculation. Table 15-2 shows the percentage of fuel pins that exceed the DNBR limit for each case.

15.12.5 Peak RCS Pressure Analysis

The peak RCS pressure for the SIMULATE-3K rod ejection accident is determined by a system analysis simulation that uses a boundary condition of the coolant expansion rate in the core. The core coolant expansion rate is calculated for each fuel assembly and is summed into a total expansion rate. The total coolant expansion rate is then input to the system analysis, which results in a pressurizer insurge and a compression of the pressurizer steam bubble. The peak RCS pressure results from the 102% power BOC case. Figure 15-36 shows the pressure transient for the Mk-B-HTP core with UO_2-Gad fuel rods.

15.12.6 Environmental Consequences

A conservative consequences analysis for a postulated rod ejection accident is performed to determine the resulting radiological consequences. The rod ejection accident calculation is based on the approach provided in Regulatory Guide 1.183. Activity is released to the environment by releases associated with the normal operation of plant equipment or the operation of plant equipment as intended in response to the accident, and as part of the subsequent cooldown activities.

Two activity release paths are evaluated separately. The first release path is via containment leakage resulting from release of activity from the primary coolant and failed fuel pins to the Reactor Building. The second path is the contribution of primary-to secondary leakage and contaminated secondary coolant release to the atmosphere. At the time of the accident, forty-five percent (45%) of the fuel rods in the core are assumed to fail due to DNB, releasing stored gap activity; no fuel melting is assumed to occur. The source term isotopic inventory is based upon fuel depletion and projected fission product inventories at the end of the cycle with the maximum thermal power uncertainty applied. An initial source term inventory is also modeled for the secondary side. The maximum Technical Specification allowed DEI concentration is modeled to be present in the secondary side water. Radioactive depletion by decay is credited during the accident.

Fission products in the fuel gap regions of fuel pins undergoing DNB are assumed to be instantaneously released to the Reactor Building atmosphere. The assumed containment leak rate is the maximum rate allowed by Technical Specifications. No credit is taken for iodine removal from the containment atmosphere by the Reactor Building sprays. Credit is taken for removal of particulates in the Reactor Building atmosphere by natural deposition.

In order to transport and release primary activity to the environment, a primary to secondary release path is modeled in the steam generators. This path is postulated to exist at the start of the accident, but is not caused by the rod ejection accident. The assumed primary to secondary steam generator tube leakage rate is the maximum rate allowed by ONS Technical Specifications.

The thermal/hydraulic model discussed in the previous sections is used as the basis for the plant response and steam releases modeled in the environmental analysis. The plant is initially operating in a normal mode at full power (plus maximum thermal power uncertainty) with primary to secondary leakage. When the break initiates, the activities in the primary and secondary side are modeled to be instantaneously and homogeneously released to their respective systems. Shortly after the initiation of the event, the reactor is automatically tripped. The steam generators are assumed to discharge activity directly to the environment. This steam header will repressurize resulting in lifting its Main Steam Relief Valves. Since the steam release from the affected steam generator is not isolable, this release will continue as long as water and conditions conducive to boiling exist in this steam generator. Plant cooldown is achieved by discharging steam directly to the environment through the Atmospheric Dump Valves (ADVs). No credit is taken for the condenser.

Since Oconee Nuclear Station is a B&W designed plant, it uses once through steam generators which provide for vertical tubing which carries primary coolant from the top of the generator to its bottom while exchanging heat with the secondary fluid on the shell side. Because of this tubing arrangement, the tube leakage is assumed to occur above the secondary water mass in the steam generator. Iodine partitioning in the steam generator is credited in accordance with Regulatory Guide 1.183 but no credit is taken for iodine plateout in the steam generator or steam lines.

When the thermodynamic conditions are met for the Low Pressure Injection (LPI) system to remove decay heat from the primary, cooldown releases from the ADVs cease and decay heat removal is accomplished by the LPI system. Primary to secondary leakage continues until the temperature of the primary water leaking is less than the boiling point for water at atmospheric conditions. Offsite atmospheric dispersion factors from the Updated Final Safety Analysis Report Chapter 2 were used.

Based upon this model, releases of activity to the environment from the primary and secondary systems can be calculated and used to calculate doses offsite at the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ) and in the Control Room. The doses calculated meet the regulatory criteria of 10 CFR50.67 for each of the source terms examined. The results are presented in Table 15-16.

15.12.7 Conclusions

The rod ejection accident is analyzed for six cases which include different initial conditions for power level, number of RCPs in operation, ejected rod worth, and core physics parameters associated with BOC and EOC conditions. For the full Mk-B-HTP core with UO_2-Gad fuel rods, Table 15-2 shows the peak fuel pellet radial average enthalpy and fuel cladding failure percentage limit, for each of the transient scenarios, and peak RCS pressure for the limiting scenario. The environmental consequences analysis results are within the 10CFR50.67 limits. All of the acceptance criteria are met.

15.12.8 References

1. Deleted per 1999 Update
2. Deleted per 1999 Update
3. Deleted per 1999 Update
4. Deleted per 1999 Update
5. Deleted per 1999 Update
6. Deleted per 1999 Update
7. Deleted per 1999 Update
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9. Deleted per 1999 Update
10. Deleted per 1999 Update
11. Deleted per 1999 Update
12. Deleted per 1996 Update
13. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components", ASME

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15.13.1 Steam Line Break Accident

15.13.2 Identification of Causes and Description

The steam line break accident is caused by a rupture of one of the two main steam lines. A spectrum of break sizes up to and including a double-ended guillotine rupture are postulated. For steam line breaks that result in reactor trip, the limiting break size is a double-ended guillotine rupture since it maximizes the cooldown of the RCS. Smaller steam line breaks that do not result in reactor trip are analyzed in Section 15.17. The expected plant response to a double-ended guillotine rupture of one the main steam lines with offsite power maintained is as follows. The break initially results in a rapid blowdown of both steam generators. The steam generator depressurization initiates a rapid Reactor Coolant System (RCS) cooldown and depressurization, which results in a reactor trip on variable low pressure-temperature within the first few seconds of the accident. The reactor trip causes the main turbine stop valves to close, thereby isolating the affected steam generator from the unaffected steam generator. The affected steam generator continues to depressurize while the unaffected steam generator repressurizes. The main feedwater (MFW) pumps are tripped, the main and startup FDW control valves on the affected steam generator are closed, and the turbine-driven emergency feedwater (EFW) pump is inhibited from starting, Automatic Feedwater Isolation System (AFIS) circuitry is actuated on low steam generator pressure. The motor-driven EFW pumps start on main feedwater pump trip. The operator will manually trip all reactor coolant pumps (RCPs) on a loss of the subcooled margin. The motor-driven EFW pump to the affected steam generator is tripped by the AFIS circuitry when the rate of depressurization setpoint is exceeded. EFW flow is automatically controlled to the unaffected steam generator to provide the secondary heat sink. The High Pressure Injection System (HPI) will actuate on low RCS pressure and will begin restoring RCS inventory. The operator will then throttle HPI flow to maintain pressurizer level to the normal post-trip level.

The steam line break accident is analyzed both with and without offsite power. The with offsite power maintained case analyzes end-of-cycle core conditions to maximize the positive reactivity addition resulting from the RCS cooldown and any resulting return-to-power. The without offsite power case analyzes beginning-of-cycle (BOC) core conditions to conservatively predict the approach to DNB as the reactor coolant pumps (RCPs) coast down. No credit is taken for the Automatic Feedwater Isolation System (AFIS) circuitry since some of the components that actuate are non-safety grade. The non-safety grade Integrated Control System (ICS) is assumed to be in manual control with no operator action, since this assumption has been demonstrated to be conservative relative to assuming ICS control of MFW. This results in uncontrolled MFW flow and actuation of the EFW System. The results presented model the replacement steam generators. The analysis methodology and the computer codes used in the analysis are given in Table 15-33. The initial conditions are given in Table 15-34. The Reactor Protective System and Engineered Safeguards Protective System setpoints and delay times are given in Table 15-35.

Operator action to isolate MFW flow to the broken steam generator is credited at 10 minutes. The limiting single failure for the with offsite power analysis is the failure of a train of engineered safeguards that results in only one train of HPI. No single failure was identified which affects the results of the without offsite power analysis. The maximum worth control rod is assumed to remain in the fully withdrawn position.

The steam line break accident is considered to be a limiting fault. The acceptance criteria for this event are that the core will remain intact for effective core cooling and that the offsite doses

will be within 100% of the 10CFR50.67 limits. The RETRAN system thermal-hydraulic analysis results are valid for the full core with Mk-B-HTP fuel.

15.13.3 With Offsite Power Analysis

The steam line break accident with offsite power analysis is concerned with the magnitude of any post-trip return-to-power. A significant return-to-power with the presence of a stuck rod may challenge the DNB limit. The limiting scenario with respect to maximizing the overcooling and reactivity addition has been determined to be the case with the ICS in manual control with no operator action, which results in uncontrolled MFW flow and actuation of the EFW System. This limiting scenario has been determined to bound scenarios with the ICS controlling MFW flow to the post- trip steam generator level setpoint increased by an allowance for uncertainty. The duration of the analysis is 10 minutes, which includes the core conditions of minimum DNB margin. The results of the analysis are shown in Figure 15-40, Figure 15-41, Figure 15-42, Figure 15-43, Figure 15-157, Figure 15-158, and Figure 15-159, and the sequence of events is given in Table 15-5.

The steam line break initially causes the pressure to decrease in both steam generators (Figure 15-40). The reactor trips in 3.1 seconds. Break flowrates (Figure 15-41) for both steam generators rapidly increase. After the turbine stop valves close, break flow from the unaffected steam generator stops. Break flow from the affected steam generator decreases with decreasing pressure, and the unaffected steam generator repressurizes until about 30 seconds. The uncontrolled main feedwater flow overfills the affected steam generator at approximately 240 seconds, and the unaffected steam generator at 214 seconds. The cooldown in the affected loop leads the cooldown in the unaffected loop, as shown in the cold leg and hot leg temperature responses (Figure 15-42). RCS has cooled to less than 250°F by the end of the simulation.

The total, moderator, Doppler, boron and control rod reactivities are presented in Figure 15-43. The negative reactivity insertion at the beginning of the transient is due to the reactor trip and control rod insertion. The cooldown causes positive reactivity insertion due to the negative moderator and Doppler coefficients. The core remains subcritical throughout the post-trip period, with the minimum subcritical margin reached at about 110 seconds. Boron injection from the core flood tanks, and then later the HPI system, provides sufficient negative reactivity to maintain the subcritical margin. The reactor power (Figure 15-157) decreases rapidly on reactor trip. The thermal power generally follows the neutron power response and then approaches the decay heat power level. The minor fluctuations in the heat flux are caused by flow surges in the core which result from flow degradation due to two-phase conditions in the unaffected loop. RCS pressure (Figure 15-158) rapidly decreases until the affected loop and reactor vessel head begin to saturate at approximately 4 seconds. After this time, RCS pressure continues to decrease for the remainder of the simulation.

Core inlet mass flow (Figure 15-159) initially increases with time due to the decreasing RCS temperatures. However, as the unaffected loop begins to void and RCP performance degrades, core inlet flow decreases to approximately 80% of the initial flow. Core flood tank and HPI System injection refill the RCS, and single phase flow is restored by 160 seconds.

Based on the reactor remaining subcritical post-trip, no return-to-power occurs. Therefore, the DNBR is bounded by the steam line break without offsite power case, and no detailed VIPRE-01 analysis is necessary.

15.13.4 Without Offsite Power Analysis

The steam line break accident without offsite power analysis assumes a loss of offsite power coincident with the break which trips the reactor and causes the RCPs to coast down. For this scenario the steam line break accident is a loss of flow accident with a coincident depressurization. The minimum DNBR statepoint occurs within the first few seconds of the RCP coastdown, therefore the duration of the analysis is 5 seconds. The results of the analysis are shown in Figure 15-161, Figure 15-162, Figure 15-163, Figure 15-164, Figure 15-165, Figure 15-166 and Figure 15-167, and the sequence of events is given in Table 15-48. The steam line break initially causes the pressure to decrease in both steam generators (Figure 15-161). Once the main turbine stop valves close, the unaffected steam generator starts to repressurize. The affected steam generator has depressurized to about 750 psig by the end of the analysis. The break flow response is similar to the offsite power analysis. The cooldown in the affected loop is almost the same as in the unaffected loop during the first 5 seconds, as shown in the cold leg temperature response (Figure 15-162). The increase in hot leg temperatures is caused by the flow coastdown. The affected loop hot leg temperature is slightly higher than the unaffected loop hot leg temperature due to the post-trip outsurge from the pressurizer. The RCS volumetric flow decreases for the duration of the simulation (Figure 15-163). The control rod insertion on loss of offsite power determines the core kinetics response (Figure 15-164). Due to the assumed BOC kinetics parameters and the short duration of the analysis, the moderator and Doppler reactivity feedback is negligible. The reactor neutron power decreases rapidly on reactor trip (Figure 15-165), with the thermal power responding slower due to the thermal delay. RCS pressure (Figure 15-166) rapidly decreases due to the effects of the overcooling from the steam line break and from the control rod insertion. As flow and primary-to-secondary heat transfer begin to degrade, RCS pressure begins to recover.

The system analysis results are input to the detailed core thermal-hydraulic analysis to determine the limiting DNBR. The transient minimum DNBR (Figure 15-167) is 1.73 for the full core with Mk-B-HTP fuel. The minimum DNBR value is greater than the design limit.

15.13.5 Environmental Consequence for the Large Steam Line Break

A conservative consequences analysis is performed for a postulated double ended break of a main steam line. This break results in an increased thermal demand on the reactor coolant system (RCS) and a rapid cooldown and positive reactivity addition from a negative temperature coefficient. This transient is not postulated to induce fuel failures, steam generator tube failures or any other failures of fission product barriers or primary system pressure boundaries, or any other pieces of equipment. Thus, the environmental consequences result from plant releases of pre-existing RCS activity transported to the secondary side by postulated steam generator tube leakage, and of pre-existing secondary activity. This activity is then released to the environment by releases associated with the normal operation of plant equipment or the operation of plant equipment as intended in response to the accident, and as part of the subsequent cooldown activities.

Two RCS source terms are examined as part of this analysis: a preaccident iodine spike and a concurrent iodine spike. The first models the maximum Dose Equivalent Iodine (DEI) activity concentration permitted by Technical Specifications for an iodine spike at full power. This preaccident spike is postulated to occur at the time of accident initiation. This source term is modeled to be released instantaneously and homogeneously such that the RCS activity is in equilibrium at the start of the accident. The second source term models a concurrent iodine spike, where the primary system transient associated with the accident causes an iodine spike in the primary system. The increase in primary coolant iodine concentration uses a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant

increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value specified in Technical Specifications. Both iodine spike source terms also bound Technical Specification limits for Dose Equivalent Xenon (DEX).

An initial source term is also modeled for the secondary side. The maximum Technical Specification allowed DEI concentration is modeled to be present in the secondary side water, the steam generators and any makeup water supplied to the unit. Thus, the secondary side is essentially modeled as an infinite source of water at the secondary side Technical Specification DEI concentration limit.

In order to transport and release primary activity to the environment, a primary to secondary release path is modeled in the steam generators. This path is postulated to exist at the start of the accident, but is not caused by the steam line break. The tube leakage into the unaffected steam generator modeled bounds the maximum allowed tube leakage rate into one steam generator. The affected steam generator is modeled with a leakage rate that bounds the maximum allowed unidentified primary to secondary leakage allowed by Technical Specifications.

The thermal/hydraulic model discussed in the previous sections is used as the basis for the plant response and steam releases modeled in the environmental analysis. The plant is initially operating in a normal mode at full power (plus maximum thermal power uncertainty) with primary to secondary leakage. The only releases occurring at the start of the accident are from the condensate steam air ejectors (CSAEs), which discharge a mixture of motive steam and condensate gases. Since the CSAEs operate continuously, no gases are assumed to be in the secondary system, as they would be removed by the CSAEs when introduced into the secondary system. When the break initiates, the activities in the primary and secondary side are modeled to be instantaneously and homogeneously released to their respective systems. Shortly after the break initiates, the reactor is automatically tripped and radioactive decay (and daughter product production) is begun in the model. The affected steam generator begins to discharge all of its activity directly to the environment. The unaffected steam generator also discharges its inventory directly to the environment through the break until the Turbine Stop Valves close shortly after reactor trip. This steam header will repressurize resulting in lifting its Main Steam Relief Valves. Since the steam release from the affected steam generator is not isolable, this release will continue as long as water and conditions conducive to boiling exist in this steam generator.

In order to maximize releases to the environment, the condenser is assumed to not be available. This requires that the unit be cooled down using the unaffected steam generator by discharging steam from this steam generator directly to the environment through the Atmospheric Dump Valves (ADVs). No credit is taken for the condenser and no partitioning credit is taken for CSAE releases which are modeled to occur until the beginning of cooldown.

The large steam line break causes the Turbine Driven Emergency Feedwater Pump (TDEFWP) to start and briefly supply makeup water. The TDEFWP is driven by steam from the Main Steam System or the Auxiliary Steam System and exhausts directly to the environment, and therefore, is a release path that is included in the environmental analysis.

Since Oconee Nuclear Station is a B&W designed plant, it uses once through steam generators which provide for vertical tubing which carries primary coolant from the top of the generator to its bottom while exchanging heat with the secondary fluid on the shell side. Because of this tubing arrangement, the tube leakage is modeled to occur above the secondary water mass in the steam generator. Therefore, no credit is taken for iodine partitioning in the steam generator. No credit is taken for iodine plateout in the steam lines or any other surface.

After the plant is stabilized following the initial transient, a soak is required. After the soak is completed, the plant is cooled down at the maximum rate permitted by Technical Specifications. This rate is reduced as required by Technical Specifications at the appropriate temperature. When the thermodynamic conditions are met for the Low Pressure Injection (LPI) system to remove decay heat from the primary, cooldown releases from the ADVs cease and decay heat removal is accomplished by the LPI system. Primary to secondary leakage and its release to the atmosphere continue until the temperature of the primary water leaking is less than the boiling point for water at atmospheric conditions. At this point all releases of activity from the plant model cease.

Offsite atmospheric dispersion factors from the Updated Final Safety Analysis Report Chapter 2 were used. Dose conversion factors from Federal Guidance Reports 11 and 12 were used.

Based upon this model, releases of activity to the environment from the primary and secondary systems can be calculated and used to calculate doses at the Exclusion Area Boundary (EAB), the Low Population Zone (LPZ), and in the Control Room. The doses calculated meet the regulatory criteria of 10 CFR 50.67 for each of the source terms examined. The results are presented in Table 15-16.

15.13.6 Conclusions

The steam line break accident has been analyzed both with and without offsite power. The results of the analysis show that DNBR margin exists. The results of the environmental consequences analyses are within the 10CFR 50.67 limits. All of the acceptance criteria are met.

15.13.7 References

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15.14.1 Loss of Coolant Accidents

15.14.2 Identification of Accidents

A failure of the RCS pressure boundary will result in a loss of primary coolant inventory and the potential for the core to uncover. These hypothetical failures are considered to occur in all piping and components up to and including a double-ended rupture of the largest pipe in the system. If the core is not rapidly reflooded and long term heat removal established, decay heat will cause the fuel cladding to fail and release the fission product inventory. The Emergency Core Cooling System (ECCS) is designed to deliver sufficient coolant to provide the necessary core decay heat removal for all credible loss-of-coolant accidents (LOCA).

15.14.3 Acceptance Criteria

In order to judge the acceptability of the performance of the ECCS in mitigating a LOCA, the Final Acceptance Criteria specified in 10CFR50.46 require that the results of the LOCA analysis meet the following criteria.

15.14.3.1 Peak Cladding Temperature

The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

15.14.3.2 Maximum Cladding Oxidation

The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

15.14.3.3 Maximum Hydrogen Generation

The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

15.14.3.4 Coolable Geometry

Calculated changes in core geometry shall be such that the core remains amenable to cooling.

15.14.3.5 Long-Term Cooling

After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Conformance with the acceptance criteria must be demonstrated in a LOCA analysis which is conducted within the guidelines of 10CFR50 Appendix K, "ECCS Evaluation Models." Appendix K outlines the assumptions and analytical methods which have been accepted by the Nuclear Regulatory Commission (NRC) for evaluating the consequences of LOCA. The ECCS evaluation model applicable to Oconee is detailed in the following section.

15.14.4 ECCS Evaluation Model

15.14.4.1 Methodology and Computer Code Description

The large break LOCA (LBLOCA) evaluation model, which has been approved by the NRC, is detailed in the topical report "BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants" (Reference 40). The LBLOCAs are analyzed with the RELAP5/MOD2-B&W computer code (Reference 38). The LBLOCA evaluation model has been shown to conform to the requirements of 10 CFR 50 Appendix K.

The RELAP5/MOD2-B&W code (Reference 38) solves the evolution of system hydrodynamics, core power generation, and clad temperature response during blowdown for the LBLOCA. The REFLOD3B code (Reference 4) is used to determine the length of the refill period and the flooding rates during reflood. The CONTEMPT code (Reference 5) calculates the Reactor Building pressure response. The BEACH code (Reference 39) is used with the output from REFLOD3B and CONTEMPT to determine the fuel thermal and mechanical response and the PCT during the reflood period. The code interfaces for the LBLOCA are shown in Figure 15-44 for cold leg breaks larger than 2 ft^2.

Cold leg break sizes between approximately 0.75 ft^2 and 2 ft^2 produce thermal-hydraulic behaviors that are transitional in nature, having both large and small break characteristics with respect to the evaluation model assumptions. The smaller break sizes result in slower transients for which no refill period exists. The smallest breaks may also begin reflooding the core shortly after core flood tank flow begins. The analysis of break sizes in this range requires adjustments to the nominal RELAP5-based LBLOCA evaluation model. These adjustments are described in Reference 40.

Hot leg breaks have many thermal-hydraulic similarities to the transitional breaks. There is no refill period due to direct venting of core steam to the break. Core reflooding begins shortly after core flood tank flow begins. Thus, the cold leg break LOCA methods are not suitable for analyzing these breaks. The techniques used to analyze the hot leg breaks with RELAP5/MOD2-B&W are described in Reference 40.

The small break LOCA (SBLOCA) evaluation model, which has been approved by the NRC, is detailed in the topical report “BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants” (Reference 40). The SBLOCA events are analyzed with the RELAP5/MOD2-B&W computer code (Reference 38). The SBLOCA evaluation model has been shown to conform to the requirements of 10CFR50 Appendix K.

15.14.4.2 Simulation Model

The RELAP5 LBLOCA nodalization is presented in Reference 40. A detailed nodalization of the primary loop and reactor vessel is included. For break locations other than the pump discharge, the nodalization is appropriately modified.

The RELAP5 SBLOCA nodalization is detailed in Reference 40. A detailed nodalization of the primary loop and reactor vessel is included. The secondary side nodalization is sufficient for modeling the effects of emergency feedwater delivery and steaming.

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15.14.4.3 Thermal Hydraulic Assumptions

Thermal hydraulic conditions and parameters are assumed in accordance with 10 CFR 50 Appendix K.

15.14.4.3.1 Sources of Heat

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The reactor is initially operating at 102 percent of 2,568 MWt, the maximum rated power for an Oconee class plant. Core peaking factors are obtained from the analysis based on the criteria of 10CFR50.46. Core stored energy and fuel temperatures are calculated using the TACO3 code (Reference 35). Fission product decay heat is given by 1.2 times the ANS standard and decay of actinides is also assumed greater than the ANS decay curve. Direct moderator heating accounts for 2.7 percent of the fission energy released during the blowdown. Metal-water reaction is calculated using the Baker-Just equation without steam limiting. Heat transfer from non-fuel sources is accounted for, as is primary to secondary heat transfer.

15.14.4.3.2 Fuel Mechanical and Thermal Response

The detailed fuel response throughout the duration of the transient is predicted by the RELAP5/MOD2-B&W and BEACH codes for large break LOCA and the RELAP5/MOD2 – B&W code for small break LOCA. Thermal expansion, elastic and plastic deformation, and the events leading to possible clad rupture are considered. Approved models for heat capacity and conductivity in the fuel, and gap conductance and heat transfer are used. Models for cladding, swelling and rupture are described in NUREG-0630 and are incorporated in Reference 40. Evaluation model changes to analyze M5 cladding material are documented in Reference 43.

15.14.4.3.3 Blowdown Model

ECCS bypass is predicted to occur as long as the flow velocity is calculated to be sufficient to carry the ECCS fluid away from the core. The end of blowdown is considered either when zero leak flow occurs or when ECCS water starts entering the core. Friction and form loss factors account for system pressure drops and compare well with measured plant data. Single-phase and two-phase pump models are derived from homologous relationships.

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Break flow is calculated using the Extended Henry-Fauske equation for qualities up to 0.0 at which time a switch to the Moody correlation occurs. A range of discharge coefficients is evaluated in the LBLOCA break spectrum analysis. The critical heat flux (CHF) correlations used are the B-HTP, BWC, BWCMV, Barnett, and modified Barnett. In the low flow regime, a combination of the MacBeth and Griffith correlations is used. Pre-CHF heat transfer uses the

maximum of the Dittus-Boelter or Rohsenow-Choi correlations for forced convection and a combination of the Chen, Thom, and Schrock-Grossman correlations for the nucleate boiling and forced convection vaporization regimes.

The post-CHF heat transfer regimes include transition boiling, film boiling, and single-phase steam heat transfer. For transition boiling, the correlation of McDonough, Milich, and King is used. The maximum of the Condie-Bengston and Rohsenow-Choi correlations is used in the film boiling regime. The single-phase heat transfer to steam correlation is the sum of a convective term and a radiation term. The convection heat transfer is the maximum of the McEligot or Rohsenow-Choi correlations. The radiation heat transfer is from the Sun correlation.

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15.14.4.3.4 Post-Blowdown Model

The evaluation of the LOCA during refill and reflood is conservatively conducted assuming the minimum containment backpressure consistent with the Reactor Building Cooling Systems performance, the ECCS injection with the design single failure, and conservative containment initial conditions, volume, and heat sink data. The REFLOD3B code calculates the heat transfer and hydraulic response with containment pressure input from CONTEMPT. During the refill period the core undergoes an adiabatic heatup. Steam venting and steam-water interaction, liquid entrainment, hot wall effects, and refill-reflood heat transfer are accounted for.

15.14.4.3.5 Availability of Reactor Coolant Pumps

Sensitivity studies have shown that for the large break LOCA the highest PCT results for the case with reactor coolant pumps (RCPs) tripped. Therefore, for large break LOCA the pumps trip and coast down on a loss of offsite power coincident with the break.

The SBLOCA has been analyzed assuming that the reactor coolant pumps trip and coast down coincident with reactor trip. This results in the coolant inventory change due to loss out the break and HPIS injection being reflected by the reactor vessel mixture level. The break size which resulted in the highest PCT was determined by a break spectrum analysis. This scenario was expected to represent the worst case SBLOCA, since if the reactor coolant pumps were running, the core would be cooled by pumping a two-phase mixture through the core, and no heatup would occur. Studies (Reference 14) have shown that for certain SBLOCAs characterized by a limited range of break sizes and break locations, that a delayed reactor coolant pump trip at high system void fractions can result in extended core uncovery and consequences in excess of the 10CFR50.46 criteria. This constituted a new worst case scenario. This situation resulted in the implementation of operating procedures which instruct the operator to trip the reactor coolant pumps upon loss of subcooled margin (Reference 15).

15.14.4.3.6 ECCS Performance and Single Failure Assumption

The ECCS is comprised of two passive core flood tanks (CFT), each of which injects through its associated core flood line into the reactor vessel downcomer; three low pressure injection pumps separated into two trains which inject into separate core flood lines; and three high pressure injection pumps separated into two trains which split and inject into each cold leg. The ECCS configuration was analyzed with the CRAFT2-based evaluation model (Reference 1) to determine the worst single failure in addition to the assumption of the loss of offsite power for each LOCA (Reference 33). Historically, the worst single failure for a LOCA is the loss of one bus of emergency power which results in the loss of one train of HPI and one train of LPI. The failure of transformer CT-4 has been identified as a more limiting single failure for the large

break LOCA. The failure of transformer CT-4 results in a longer delay until delivery of ECCS fluid to the RCS. However, two ECCS trains are available with this single failure. Reference 33 demonstrates that having two ECCS trains injecting at a later time is more limiting than having one ECCS train injecting at an earlier time.

The Keowee hydro unit will start up and accelerate to full speed in 23 seconds or less (Section 6.3.3.3). The failure of transformer CT-4 results in an additional 10 second delay before power is available to the ECCS pumps. The time delay between breaker closure and valve/pump motors operating at rated voltage/speed is 5 seconds. Thus, for the large break LOCA analyses performed with the RELAP5-based evaluation model (Reference 40), the LPI valves will begin to open at 38 seconds with a stroke time of 36 seconds or less. Credit is taken in the analysis for flow through the LPI valves while the valves are traveling to their full open position. Full LPI flow will be obtained within 74 seconds. Two ECCS trains are available with the single failure of transformer CT-4. However, only one train of LPI flow is credited in the actual large break LOCA analyses (Reference 42).

For the limiting large break LOCA, the core heatup following blowdown is mitigated by core flood tank injection. Typically the time of PCT is prior to the actuation of pumped ECCS flow from the LPI and HPI pumps. Flow from one LPI pump provides for the long-term cooling of the core. For smaller large break LOCAs down to the transition break size, some HPI flow contributes to core cooling prior to the time of PCT, but it is a small contribution relative to the core cooling provided by the core flood tanks and the LPI pump. The PCTs for the smaller large break LOCAs have a large margin to the 2200°F acceptance criterion, and the small contribution of HPI flow to core cooling is not significant. Therefore HPI pumps are not required for large break LOCA mitigation.

A SBLOCA does not progress as rapidly as a large break LOCA. Thus, for a SBLOCA, the timing of ECCS injection is not as significant as with a large break LOCA. For this reason, the worst single failure for a SBLOCA remains the loss of one bus of emergency power. With the selection of an adverse break location, one half of the available HPI train would inject into the broken loop. With these assumptions the ECCS is reduced to the two CFTs, one LPI train, and one half of one HPI train. The SBLOCA analyses assume a 48 second delay until full ECCS flow is delivered to the RCS.

For the SBLOCA which does not depressurize to below the core flood tank setpoint (600 psig), only one half of one HPI train was available if the break is assumed to be in the cold leg pump discharge. This was identified as an unacceptable scenario (Reference 16). In order to deliver the required HPIS flow of 350 gal/min at 600 psig (Reference 17), the HPIS was modified to allow cross connecting of the pump discharges in order to balance the flow from two HPI pumps into the four injection locations (Reference 18, 19). This manual realignment of the HPIS is assumed to be completed within ten minutes of HPIS actuation.

The performance of the ECCS is also evaluated assuming that one of the three HPI pumps is initially unavailable. The limiting single failure leaves only one HPI pump available to inject following a SBLOCA. With only one HPI pump operating, the realignment to cross connect the pump discharges, described above, cannot be performed as a result of pump runout concerns at low primary system pressure. Significantly less HPI flow capacity results, and the power level must be reduced to 75% full power for the SBLOCA analyses to meet the acceptance criteria.

15.14.5 LOCA Analyses

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15.14.5.1 Large Break LOCA

Large break LOCA (LBLOCA) accidents can be treated analytically in three separate phases: blowdown, refill, and reflood.

The blowdown phase is characterized by the rapid depressurization of the Reactor Coolant System to a condition nearly in pressure equilibrium with its containment surroundings. Break flow is calculated using the Extended Henry-Fauske equation for qualities up to 0.0 at which time a switch to the Moody correlation occurs. A range of discharge coefficients is evaluated in the LBLOCA break spectrum analysis. Core flow is variable and dependent on the nature, size, and location of the break. Departure from nucleate boiling (DNB) is calculated to occur very quickly, at the higher power locations, and core cooling is by a film boiling process. Since film boiling is only capable of removing a limited amount of heat, the cladding temperature may increase up to ~1000°F at the peak power location. Core flood tank (CFT) flow begins after the RCS depressurizes below the CFT fill pressure. Steam condensation caused by the CFT liquid aids the negative core flows that reduce the fuel pin temperatures during the middle blowdown period. During the last phase of blowdown, cooling is by convection to steam, and the cladding temperature begins to rise again.

The end of blowdown is considered to have occurred either when zero leak flow occurs or the ECCS water starts to enter the core. ECCS bypass is predicted to occur when the flow velocity is calculated to be sufficient to carry the ECCS fluid away from the core.

Following blowdown, a period of time is required for the CFTs to refill the bottom of the reactor vessel before reflood and final core recovery can be established. During this period, core cooling is marginal and the cladding experiences a near-adiabatic heatup. This period is designated as the refill phase, because the CFT flow is refilling the reactor vessel lower plenum.

When the water level reaches the bottom of the active fuel the reflood phase begins. Core cooling is by steam generated below the rising water level. The cladding temperature excursion is generally terminated before a particular elevation is covered by water since the steam-water mixture is sufficient to remove the relatively low decay heat being generated at this time. A two-phase mixture eventually covers the core, and the path to long-term cooling is established through initiation of Low Pressure Injection (LPI) System flow near the time the CFTs empty and subsequent operator action to maintain pumped injection.

The evaluation of the LOCA during refill and reflood is conservatively conducted assuming the minimum containment back pressure consistent with the Reactor Building Cooling System performance, the ECCS injection with designed single failure, and conservative containment initial conditions, volume, and heat sink data. The REFLOD3B computer code (Reference 4) calculates the heat transfer and hydraulic response with containment pressure input from the CONTEMPT computer code (Reference 5). The containment pressure used in the Oconee large break LOCA analysis is presented in Figure 15-177.

15.14.5.1.1 Large Break LOCA Break Spectrum

Using the CRAFT2-based evaluation model (Reference 1), a spectrum of large breaks was analyzed for both double-ended and longitudinal split breaks in all locations. The methodology used to identify the worst break was as follows. A double-ended break with discharge coefficient C_D = 1.0 was analyzed at the hot leg, cold leg pump suction, and pump discharge. The cold leg pump discharge was determined to be the worst break location. The break size was then varied for both double-ended and split breaks.

The RELAP5 large break LOCA analyses have replaced the CRAFT2 large break analyses. The generic break spectrum studies performed with the RELAP5 evaluation models have

selected the transition break size to be 0.75 ft^2, based on the onset of the occurrence of early cladding DNB during the blowdown phase. Both of these break spectrum studies have shown that the phenomena in the transition break size range are predicted to be similar, and that the PCTs in the vicinity of the transition break size are non-limiting.

The break spectrum analysis was also performed using the RELAP5-based evaluation model for the generic raised loop design (Reference 40) for break sizes ranging from 0.75 ft^2 up to and including the cross sectional area of the largest pipe in the system. Breaks that were clearly shown to be non-limiting in the generic break spectrum analysis were not reanalyzed for the Oconee-specific break spectrum. A double-ended break with discharge coefficient C_D = 1.0 was analyzed at the cold leg pump discharge and cold leg pump suction in the Oconee-specific break spectrum. The cold leg pump discharge was determined to be the worst break location. This break location was further analyzed for a double-ended break with discharge coefficients of C_D = 0.8 and C_D = 0.6. A split break at the cold leg pump discharge was also analyzed. The results of these analyses are shown in Table 15-6 and Figure 15-50. A symmetric power shape with an axial peaking factor of 1.7 and a peak linear heat rate of 17.5 kW/ft is assumed.

The worst break was identified as the double-ended cold leg break at the pump discharge with C_D = 1.0. Using the RELAP5-based evaluation model (Reference 40), this break of 8.55 ft^2 area yielded a predicted PCT of 1957°F and a maximum local metal-water reaction of 2.02 percent. The same break size at the pump suction showed a predicted PCT of 1830°F and a maximum local metal-water reaction of 1.54 percent. The range of break sizes smaller than the full area double-ended break at the pump discharge all showed less severe consequences.

A series of large breaks are analyzed from an initial condition where three reactor coolant pumps are in operation. Three possible break locations associated with this mode of operation were identified.

An evaluation was made using the RELAP5-based evaluation model on a generic basis for a raised-loop plant (Reference 40). Breaks were analyzed with the idle pump simulated in the intact loop, broken leg, and intact leg of the broken loop. The case with the idle pump in the broken leg was determine to be limiting. Thus, a double-ended break with the idle pump in the broken leg and a C_D = 1.0 was analyzed for Oconee using the RELAP5-based evaluation model with three pumps. This analysis, which was performed at 80 percent FP with a moderator temperature coefficient of +1 pcm/°F, was shown to be less limiting than the 100 percent FP case with a moderator temperature coefficient of 0 pcm/°F (Reference 42).

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15.14.5.1.5 Full Core Mark-B-HTP Large Break LOCA Linear Heat Rate Limits

Beginning with Oconee Unit 2 Cycle 26, Oconee core designs will consist of a full-core of Mark-B-HTP fuel assemblies, incorporating gadolinia as an integral burnable neutron absorber, operating on 24-month fuel cycles. To support this fuel transition, new LOCA analyses were performed to determine linear heat rate (LHR) limits and corresponding PCT for the Mark-B-HTP fuel assembly with gadolinia in a full-core configuration.

The limiting break identified in the break spectrum analysis (a double-ended pump discharge break with a CD = 1.0) was used to analyze the limiting linear heat rate limits for Oconee in

accordance with the LOCA evaluation model described in Reference 40. The core model is separated into a hot pin, hot channel, and average channel as documented in Reference 38. The Oconee-specific RELAP5/MOD2-B&W model was used, including the replacement once-through steam generators (ROTSG) and the passive LPI cross connect modification (Reference 50).

In addition to the Oconee input model changes made to reflect the ROTSG and the passive LPI cross connect modification, another evaluation model change is a result of resolution of Preliminary Safety Concern (PSC) 1-99. It was determined for Oconee (Reference 44) that a minimum two-phase RCP degradation model produced more limiting results than maximum pump degradation. This model assumption is different than that presented in Reference 40. Since the minimum two-phase pump degradation model produced more limiting results, all Oconee LOCA limit calculations use this model assumption.

Using this model, LOCA linear heat rate limits were determined for the Mark-B-HTP fuel assembly design. Specific calculations were performed to simulate five axial power peaks centered at the middle of the five grid spans (at core elevations of 2.506, 4.264, 6.021, 7.779, and 9.536 feet). These cases were analyzed with an axial peak of 1.7 and the radial peak was adjusted to obtain an allowable LHR limit. The initial fuel conditions for the desired peaking conditions are obtained from the TACO3 fuel performance code (Reference 35).

Calculations are performed for all five elevations for the beginning-of-life (BOL) and middle-of-life (MOL) conditions. The results of the BOL LOCA limits analyses are tabulated in Table 15-62 for UO_2 fuel, and Table 15-63 for fuel with UO_2-gadolinia fuel rods. Plant operation within these LHR limits assures that the 10CFR 50.46 acceptance criteria are not exceeded. In addition, the results for the 2.506 foot elevation at BOL conditions are presented in Figure 15-219, Figure 15-220, Figure 15-221, Figure 15-222, Figure 15-223, Figure 15-224, Figure 15-225 and Figure 15-226. These figures are representative of the results that are seen at all core elevations and times in life. These results indicate a maximum PCT of 1913.2 °F, a maximum local oxidation of less than 2.4 percent, and a whole core hydrogen generation of less than 0.16 % for a full-core of Mark-B-HTP fuel.

The gadolinia fuel has a lower fuel thermal conductivity and volumetric heat capacities than the UO_2 fuel, and therefore will respond more slowly to changes in the thermal environment. These small property differences are accounted for by reducing the LHR limits for gadolinia to keep the calculated results for gadolinia pins similar to the UO_2 results. The gadolinia pins were analyzed for LHR limits at the 2.506 foot core elevation for BOL, MOL, and EOL conditions.

The end-of-life (EOL) UO_2 LHR limits were established at the design rod average burnup of 62 GWd/mtU. However, at EOL, the TACO3 LOCA initialization is limited to a LHR that achieves a maximum initial pin pressure, because it is generally not limited by the LOCA PCT. One representative LBLOCA analysis at the 2.506 foot elevation is performed to confirm that EOL is not PCT limited

15.14.5.1.6 Full Core Mark-B-HTP Large Break LOCA Reanalysis for Error Corrections

The base LBLOCA analysis for Oconee with full-core Mk-B-HTP fuel currently described in Section 15.14.4.1.5 contained an emergency core cooling system (ECCS) end of bypass timing error, and the B&W plant ECCS evaluation model (Reference 40) did not include a column weldment model in the reactor vessel upper plenum. The estimated impact of these errors on

peak cladding temperature (PCT) was reported to the NRC in accordance with 10CFR 50.46 by Duke Energy in Reference 52.

Subsequent to Duke Energy's reporting of the ECCS end of bypass timing and column weldment errors, AREVA identified another error in the base LBLOCA analysis for Oconee for a control variable in the RELAP5/MOD2-B&W blowdown model. This control variable is used for the calculation of the core flows using a low pass filter with the variable filter break frequency supplied by the control variable. These errors do not affect the Small Break Loss of Coolant Accident (SBLOCA) analysis for Oconee applicable to a full-core of Mark-B-HTP fuel.

To address these three errors, AREVA developed a detailed column weldment model located over the top of the hot fuel channel for use in new LOCA analysis. AREVA has performed an Oconee LBLOCA reanalyses for full core Mark-B-HTP fuel to incorporate all error corrections, specifically the ECCS bypass timing error, column weldment modeling, and the filtered flow variable. The results of the LBLOCA reanalysis with respect to PCT have been reported to the NRC in accordance with 10CFR 50.46 in Reference 53.

The impact of all three errors was evaluated by reanalyzing three different LBLOCA scenarios for the UO_2 fuel, specifically the limiting beginning of life (BOL) case with an axial power shape peaked at the 2.506 foot core elevation, the limiting middle of life (MOL) 2.506-ft case and the limiting unruptured node 7.779-ft BOL case. In addition, the limiting gadolinia cases at BOL and MOL at the 2.506-ft core elevation were reanalyzed. For all of the reanalyzed cases, the resulting PCT was less than the base LBLOCA analyses at the allowed LOCA linear heat rate limits. The reanalyzed cases were used to develop estimated changes (reductions) to the PCTs for all other elevations at BOL and MOL for UO_2 and gadolinia fuel. The impact to the end of life (EOL) analyses were conservatively estimated with zero PCT change, as the PCT results from the base LBLOCA analyses at EOL are relatively low. Peak cladding temperature results from the reanalysis are tabulated in Table 15-67 for UO_2 fuel, and Table 15-68 for fuel with UO_2-gadolinia fuel rods.

For the LBLOCA scenarios that were explicitly reanalyzed for the error corrections, the results for maximum local oxidation and whole core hydrogen generation were less than the base LBLOCA analysis currently described in Section 15.14.4.1.5, as summarized in Tables 15-62 and 15-63. Therefore, the base LBLOCA analysis currently described in Section 15.14.4.1.5 remains the LBLOCA analysis of record for maximum local oxidation and whole core hydrogen generation.

AREVA has evaluated the effect on initial fuel temperatures due to burnup-dependent fuel pellet thermal conductivity degradation (TCD) using the fuel performance code COPERNIC2 which explicitly accounts for fuel pellet thermal conductivity degradation. AREVA has determined that the middle-of-life (MOL) and end-of-life (EOL) initial fuel temperature predictions using COPERNIC2 are significantly higher than values calculated for a similar set of conditions using the fuel performance codes TACO3 and GDTACO, which are currently a part of the approved LOCA evaluation model for B&W plants per Reference 40. The TACO3 and GDTACO fuel performance codes do not model fuel pellet thermal conductivity degradation. AREVA's assessment of this change to the initial fuel temperature results in a peak cladding temperature (PCT) increase of 428°F for the limiting MOL case, if no other actions are taken.

The second reported PCT change is due to a change (reduction) in the input values for allowable linear heat rates (LHR) used in the LBLOCA analysis. The allowable LHR values at MOL were penalized (reduced) by 2 kW/ft at all core elevations in order to maintain the MOL initial fuel temperatures at or below the fuel temperatures predicted by TACO3 and GDTACO in the current LBLOCA analsyes of record. AREVA has estimated the impact of the LHR penalty as a reduction in PCT of -428°F for the limiting MOL case. Therefore, these two reported PCT

changes offset each other, resulting in no net change to the PCT results for the Oconee LBLOCA analysis of record previously reported in Reference 53. The LHR penalty of 2 kW/ft at MOL for all core elevations is indicated via a footnote in UFSAR Tables 15-62 and 15-63. The estimated changes in PCT due to the impact of fuel pellet thermal conductivity degradation on initial fuel temperatures, and the offsetting penalty to MOL allowable LHR values, is reflected in Section 15.14.4.4 for the LBLOCA analyses. The estimated changes to LBLOCA peak cladding temperatures due to fuel pellet thermal conductivity degradation have been reported to the NRC by Duke Energy in Reference 54 in accordance with 10CFR 50.46.

15.14.5.2 Small Break LOCA and Break Spectrum Analysis

The transient progression for SBLOCAs is summarized here to identify the key phenomena and controlling thermal-hydraulic behavior during each phase of the event. A potentially limiting SBLOCA generally progresses through five phases: (1) subcooled depressurization, (2) reactor coolant pump and loop flow coastdown and natural circulation, (3) loop draining, (4) boiling pot, and (5) refill and long-term cooling. The subcooled depressurization phase begins at the leak initiation. This phase is characterized by the period of time before the RCS begins to saturate and voids begin to form in the RV upper head and hot leg U-bends. During this period, the pressurizer will begin to empty, the RCS will depressurize to the low RCS pressure reactor trip setpoint, and the turbine will trip. With the assumption of a loss of off-site power coincident with reactor trip, the MFW pumps and RC pumps will trip and EFW will be initiated following a 69-second delay.

Following the RCP coastdown, the RCS flow tends to evolve to a natural circulation flow condition. The energy generated by the core is transferred by convection to the steam generators during the flow phase. The continued loss of the RCS liquid inventory allows steam voids to form in the upper reactor vessel head and the upper hot leg U-bends. Natural circulation ends when the U-bend steam void displaces the hot leg mixture levels below the U-bend spillover elevation. Flow is usually interrupted first in the hot leg containing the pressurizer surge line connection, because of the additional flashing of the saturated pressurized liquid that enters during the subcooled depressurization. Near the end of the flow phase, alternating periods of RCS repressurization can cause intermittent spillovers of hot-leg liquid into the steam generator primary region.

With the interruption of the RCS loop flow, the loop-draining phase begins. As the entire RCS approaches saturated conditions, the onset of subcooled and saturated nucleate boiling occurs in the core because of the high decay heat levels and the RCS depressurization. The flashing within the hot legs increases the size of the voids in the U-bends and eventually interrupts RCS flow and decreases the primary-to-secondary heat transfer. For the larger SBLOCAs, the RCS will continue to depressurize as the loops drain. For smaller breaks, however, the reduced heat transfer can interrupt the RCS depressurization. Also for these smaller breaks, the volumetric expansion of the RCS, due to continued steam formation, can exceed the volumetric discharge from the break, causing the RCS pressure to temporarily stabilize or even increase.

In the reactor vessel, the steam void in the upper head displaces enough liquid to uncover the reactor vessel vent valves (RVVVs), creating a manometric imbalance between the core and the downcomer. The imbalance forces the RVVVs to open and pass steam into the reactor vessel downcomer. The downcomer steam volume grows until the cold leg nozzle is exposed to steam. As soon as the downcomer liquid level decreases below the cold leg nozzle spillunder elevation, a steam venting path develops from the core through the RVVVs to the cold leg break, enhancing the RCS depressurization.

During the loop draining phase, the steam voids that developed in the U-bends can become large enough that the primary liquid level is displaced into the steam generator tube region below the EFW nozzles. If feedwater (MFW or EFW) is injecting through the EFW nozzles, improved primary-to-secondary heat transfer can then be restored through condensation on the tubes wetted by the feedwater. This heat transfer process within a once through steam generator (OTSG) is referred to as boiler-condenser mode (BCM) cooling. When BCM cooling takes place near the location of the EFW nozzles, it is referred to as high-elevation BCM cooling. If high-elevation BCM occurs, the RCS depressurization rate will be increased. Later in the loop draining phase, a different form of BCM cooling can occur if the RCS tube liquid level decreases below the secondary liquid level. This cooling process is referred to as pool BCM cooling, and will continue if (1) RCS condensation and ECCS injection do not cause the RCS liquid level to increase above the secondary level and, (2) the secondary fluid temperature is maintained below the temperature of the steam on the primary side of the OTSG tubes. Further, if the secondary liquid level is several feet above the RCP spillover elevation then the condensate formed during this process augment the ECCS flow to the core. For the smaller breaks, the combination of leak flow (with upper-RV venting through the RVVVs), BCM cooling, and HPI cooling will cause the RCS pressure to decrease.

Also during the loop draining phase, the reactor vessel outlet annulus mixture level will decrease to the hot leg nozzle spillunder elevation. If the top of the hot leg nozzles void, steam will flow up the hot leg riser section, and liquid from the hot leg risers will drain back into the vessel. This hot leg draining allows the mixture level in the outlet annulus to remain near the top of the hot leg nozzle until the hot leg liquid level drops into the RV exit nozzle horizontal piping.

After the hot legs empty, another path for the direct venting of steam to the break can be opened if the loop seals in the RCP suction piping are cleared. Depending on the break size, the RCS depressurization can be rapid enough to cause significant flashing in the suction piping, causing the liquid level to decrease below the suction piping spillunder elevation. The loop seals will then be clear, creating another steam relief path, in addition to the path through the RVVVs.

When loop draining ends, the break site void fraction will be based on core steam plus broken loop HPI flow. At that point, the only RCS liquid available for core cooling is the liquid remaining in the reactor vessel and the ECCS flow plus any SG condensate from the intact loops if the loop seal has not cleared. This portion of the transient is defined as the “boiling pot” phase. The increased void fraction at the break will further increase the RCS depressurization rate. The reactor vessel levels will continue to decrease; however, if the ECCS injection plus SG condensate cannot match the reactor vessel liquid loss from flashing, decay heat, and passive metal heat.

The break flow allows the RCS to continue to depressurize. Once the CFT or the HPI flow rate exceeds the break discharge rate, the RCS will refill to the break elevation. Before either of these conditions occurs, the mixture levels may descend into the core heated region resulting in a heatup of the fuel cladding in the uncovered portion of the core.

The clad temperature increases calculated for the upper core elevations are conservative because a power shape skewed to the core exit is used. The peak power occurs at the 9.536-ft core elevation. This power shape bounds the positive imbalance limits at the limits of normal operation. During the period of partial core uncovering, the clad may swell and possibly rupture if the clad temperatures exceed 1300 F. The potential for clad rupture is increased in the SBLOCA analytical model by assuming an initial internal pin pressure typical of the end of fuel life (EOL). If clad rupture is calculated, a sensitivity study is needed to show that the calculated PCT will bound the fuel pin conditions at any time-in-life condition.

An SBLOCA transient analysis is normally terminated at some point after the entire core is refilled and the cladding temperatures returned to within a few degrees of RCS saturation temperature. For the level to increase, core inflow (ECCS plus SG condensate) must exceed the liquid loss rate. Continued RCS depressurization permits higher ECCS injection rates that hastens core refill. The additional ECCS flow assures that the core can be kept covered. Once the core has been completely quenched, the analytical results are checked to ensure a path to long-term cooling is established. For long-term cooling to be assured, the HPI flow and/or LPI flow must match core boiling due to decay heat and wall metal heat plus flashing. When long-term cooling is assured, the LOCA analysis is terminated.

The SBLOCA is considered to be those break sizes greater than the normal makeup capacity and less than 0.75 ft^2. The minimum size corresponds to a break size of approximately 0.0008 ft^2 with letdown flow isolated or 0.0004 ft^2 assuming normal letdown. Break locations in both the cold leg pump suction and discharge piping are considered, along with a spectrum of break sizes (0.07, 0.1, 0.125, 0.15, 0.175, 0.2, 0.3, 0.5, and 0.75 ft^2). Breaks between 0.50 and 0.75 ft^2 are part of the Mark-B11 spectrum only. Mk-B-HTP break sizes greater than 0.50 ft^2 are considered part of the LBLOCA spectrum. This approach ensures that the limiting case is identified. In addition, two special cases are analyzed. These are the 0.44 ft^2 core flood line break, and the 0.025 ft^2 HPI injection line break. These two cases are unique due to the different fraction of the ECCS flow that can spill out of the break and not contribute to core cooling. Breaks at the connection of the HPI injection line to the cold leg are limited in size to the injection line itself. A larger break at this location, which would be a nozzle break, is not required per the NRC-approved evaluation model.

The SBLOCA analyses have demonstrated that the ECCS supplies sufficient emergency coolant injection to meet the 10CFR50.46 acceptance criteria for all SBLOCAs. The HPI flow rates assumed in the core flood line, pump discharge, and HPI line break analyses are shown in Tables 15-28, 15-29, and 15-30, respectively. To address the possibility of spilling HPI water for cold leg pump discharge breaks and HPI line breaks, credit is taken in the analyses for realigning the HPI system by opening valves HP-409 and/or HP-410 within 10 minutes after ES actuation.

The SBLOCA analyses assume that the operator manually controls the Emergency Feedwater System to raise the steam generator levels to the loss of subcooled margin setpoint. Operator action to begin raising levels to the loss of subcooling margin setpoint, which enhances primary-to-secondary heat transfer, is credited starting at 20 minutes for one steam generator, and 30 minutes for the second steam generator. For all SBLOCAs below a break size of 0.06 ft^2, credit is also taken for the operator to manually steam the steam generators at 60 minutes. This action is very effective in cooling and depressurizing the primary, decreasing break flow, and increasing ECCS flow. The normal method of steaming the steam generators in remotely using the Turbine Bypass System. The analysis credits steaming the steam generators locally using the atmospheric dump valves.

15.14.5.2.1 Deleted Per 2014 Update

15.14.5.2.2 Deleted Per 2014 Update

15.14.5.2.3 Full Core Mark-B-HTP SBLOCA and Break Spectrum Analysis

A full-break spectrum was analyzed to ensure that the limiting case was appropriately determined for the full-core Mark-B-HTP configuration. A total of 17 separate break sizes were analyzed for the SBLOCA full-break spectrum. These include the 0.01, 0.04, 0.07, 0.1, 0.125,

0.15, 0.175, 0.2, 0.3, 0.4, and 0.5 ft^2 CLPD pipe breaks with LOOP. If offsite power remains available, as considered in PSC 2-00 (References 45, 46, and 47), there are break sizes that can produce an increase in cladding temperature with a manual two-minute RCP trip compared to the LOOP assumption, therefore the 0.3, 0.4, and 0.5 ft^2 CLPD pipe breaks with a manual RCP trip two minutes after reaching the loss of subcooling margin (LSCM) setpoint were analyzed. Also, a 0.02464 ft^2 HPI line break with LOOP and the 0.44 ft^2 CFT line break (with LOOP and 2-minute RCP trip) were also analyzed.

Gadolinia fuel has lower fuel thermal conductivity and volumetric heat capacities than the UO_2 fuel. The allowed LHR limits for gadolinia are reduced to control the LBLOCA PCTs. The reduction in LHR limits for gadolinia is larger than the volumetric heat capacity differences between gadolinia and UO_2. Since the LHR limit reduction for gadolinia is greater than the volumetric heat capacity ratio, the PCTs for gadolinia rods will be lower, so they are not explicitly included in the SBLOCA analyses.

A new consideration regarding axial power shapes was developed while performing scoping studies for the full-core Mark-B-HTP SBLOCA analyses. The potential of extended core uncovery was called to question for the bounding nature of the EM axial power shapes as described in the LOCA evaluation model (Reference 40). It was found that the location for the most bounding axial power shape with a peaking factor of 1.7 for any time during the cycle is now found to be 11-ft (Reference 51). Therefore, the Oconee Mark-B-HTP full-core SBLOCA analyses use a top-skewed end-of-cycle 11-ft axial power shape peaked at the 11-ft core elevation. This top-skewed axial power shape maximizes the cladding temperature increase during the time of core uncovering.

The results for the full-core Mark-B-HTP SBLOCA break spectrum at 102% of 2568 MWt are summarized in Table 15-64 and Figure 15-227. The limiting break is a 0.15 ft^2 break at the cold leg pump discharge, with a peak cladding temperature of 1597.5°F and a maximum local oxidation of less than 1.0 percent. The transient results for this limiting case are provided in Figure 15-228, Figure 15-229, Figure 15-230, Figure 15-231, and Figure 15-232.

15.14.5.2.4 Partial-Power SBLOCA Analysis

SBLOCA analyses are also performed assuming that one of the three HPI pumps is initially unavailable, and that a single failure leaves only one pump available for credit in the analysis. In this situation there is the potential for a significant fraction of the HPI flow to be spilled out of the break. The realignment of the HPI System described above cannot be performed with only one HPI pump operating. For the limiting break sizes and locations, the available HPI flow is only capable of cooling the core for initial power levels of up to 50% full power (analysis value of 52% FP). These analyses also assume that the operator raises the steam generator levels to the loss of subcooled margin setpoint as described above. Steaming of the steam generators at 25 minutes using the atmospheric dump valves is also credited.

A spectrum of potentially limiting break sizes and locations were also analyzed to determine the limiting PCT at 52% of 2568 MWt considering a full-core of Mark-B-HTP . The limiting breaks considered in the analyses were: 0.01, 0.04, 0.06, 0.07, 0.072, 0.08, 0.10, 0.13, 0.20, and 0.40 ft^2 CLPD pipe breaks considering LOOP coincident with reactor trip. If off-site power remains available, as considered in PSC 2-00 (References 45, 46, and 47), the analyses considered CLPD break sizes of 0.3, 0.4 and 0.5 ft^2 with manual reactor coolant pump trip two minutes after LSCM. Other cases considered include a 0.02464 ft^2 HPI line break and a 0.44 ft^2 CFT line break (Reference 55).

The results for the SBLOCA break spectrum at 52% are summarized in Figure 15-213. The limiting break was determined to be a 0.072 ft² CLPD break, with a PCT of 1480.2°F and a maximum local oxidation of 0.44 percent. The transient results for this case are shown in Figure 15-214, Figure 15-215, Figure 15-216, Figure 15-217, and Figure 15-218.

15.14.5.3 Evaluation of Reduced T_{ave} Operation

An analysis was performed to assess the condition under which an end-of-cycle (EOC) T_{ave} reduction could be performed. The reduced T_{ave} LBLOCA analysis was completed at 102% of 2568 MWt at the 2.506 foot elevation with an RCS temperature of 567 °F, which is the nominal RCS T_{ave} reduced by 12 °F (10 °F reduction with a 2 °F uncertainty). Using a moderator temperature feedback table based on a –10 pcm/°F, the results showed that the fuel and cladding temperature response at or near the peak power elevation are lower than in the nominal T_{ave} analysis. Therefore, an EOC T_{ave} reduction of up to 10 °F is acceptable with respect to the LOCA analysis provided the MTC is more negative than –10 pcm/°F.

15.14.5.4 10 CFR 50.46 Reporting Summary

In addition to the LOCA analyses presented in Subsection 15.14.4.1 and 15.14.4.2, LOCA evaluations may be performed as needed to address evaluation model changes or errors, or to support plant changes that affect the LOCA analysis of record. The errors or changes are evaluated, and the impact on the peak cladding temperature (PCT) is determined. The resultant increase or decrease in PCT is added to the analysis of record PCT. 10 CFR 50.46 allows for the estimates of errors in, or changes to, an ECCS evaluation model or its application. These PCT changes for the limiting transient are reported to the NRC, in accordance with requirements of 10 CFR 50.46.

For the Oconee Large Break LOCA analysis for full-core Mark-B-HTP fuel, the analysis of record as described in Subsection 15.14.4.1 has a PCT value of 1851.9°F. For 10 CFR 50.46 reporting purposes, this is rounded up to 1852°F. For the Small Break LOCA analysis for full-core Mark-B-HTP fuel, the analysis of record as described in Subsection 15.14.4.2 has a PCT value of 1597.5°F, which is rounded up to 1598°F for 10 CFR 50.46 reporting purposes. Other assessments for PCT impacts due to ECCS evaluation model changes or errors for the limiting transients are listed below, consistent with the Oconee 10 CFR 50.46 reporting summary per References 53 and 54.

Oconee Large Break LOCA Analysis of Record PCT:	1852°F
LBLOCA PCT Assessments:	
• LBLOCA PCT Assessment for Fuel Pellet Thermal Conductivity Degradation (transient effects)	+2°F
• PCT increase due to higher initial fuel average temperatures when fue pellet thermal conductivity degradation is considered	+428°F
• PCT decrease due to MOL linear heat rate penalty of 2 kW/ft at all core evaluations.	-428°F
Oconee Large Break LOCA Licensing Basis PCT for 10 CFR 50.46 Reporting:	1854°F

15.14.6 Evaluation of Non-Fuel Core Component Structural Response

The temperature transient in the core can produce significantly higher than normal temperatures in components other than fuel rods. Therefore a possibility of eutectic formation between dissimilar core materials exists. Considering the general area of eutectic formation in the entire core and reactor vessel internals, the following dissimilar metals are present, with major elements being in the approximate proportions shown:

Deleted Per 2013 Update.

Control Rod Poison Material

80% silver

15% indium

5% cadmium

Zircaloy-4

98% zirconium

1-3/4% tin

M5

99% zirconium

1% niobium

Inconel 625

58% nickel

21.5% chromium

9% molybdenum

5% iron

3.65% Nb-Ta

0.5% silicon

0.5% manganese

0.4% titanium

0.4% aluminum

Inconel 718

53% nickel

19% chromium

3% molybdenum

5% Nb-Ta

1% titanium

0.5 % aluminum

remainder iron

All these alloys have relatively high melting points (≥ 2,300°F) except those for silver, cadmium, and indium. The melting point of the silver-indium-cadmium alloy is about 1,470°F.

The binary phase diagram indicates that zirconium in the proportion 75 to 80 percent has a eutectic point with either iron, nickel, or chromium at temperatures of approximately 1,710, 1,760, and 2,380°F, respectively. If these dissimilar metals are in contact and if those eutectic points are reached, then the materials could theoretically melt even though the temperature is below the melting point of either material taken singly.

The Mk-B10 through Mk-B11A use Zircaloy-4, rather than inconel, for the intermediate spacer grids. Only the end grids are made of inconel and these grids are outside of the active fuel region. The Mk-B-HTP fuel design uses M5 material for the fuel cladding, guide tubes, and the intermediate and top spacer grids. Only the bottom spacer grid and end fittings are made from Inconel 718. Therefore, the current assembly designs are less susceptible to this phenomenon than older designs, which had inconel grids at each location.

B&W conducted experimental tests in which specimens of Zircaloy-4 tubing in contact with sections of INCONEL 718 spacer grids material were subjected to a thermal transient closely approximately that of the clad hot spot following a LOCA. These tests verified that the eutectic reaction is limited to the small region of contact between the clad and the spacer grid tips (dimples), and that it terminates as these materials melt at the point of contact. Both the clad and the grid material maintained their structural integrity because the amount of material involved was small and melting was localized.

Another area of dissimilar metal contact is that of a zirconium or M5 guide tube with the stainless steel cladding of the control rod. As noted in UFSAR Section 4.5.2.2, the Oconee units use the extended life control rod assembly (ELCRA) design which uses Inconel 625 as the cladding material. To determine whether the temperatures in the control rod following a LOCA could become high enough to approach either the temperature required for possible eutectic formation between the clad and the guide tube or the melting temperature of the Ag-In-Cd alloy, the thermal performance of a control rod assembly following a LOCA was examined analytically.

AREVA has performed a generic post-LOCA control rod survivability analysis to support all 177 fuel assembly B&W plants. The analyses for control rod integrity model the entire active length of the control rod, guide tube and annular flow channel between the control rod and the inside of the guide tube. The features are added to the RELAP5 model used in the ECCS evaluation model approved by the NRC, BAW-10192PA [Reference 40], which is performed using the RELAP5/MOD2-B&W code [Reference 38].

The control rod survivability analyses use a temperature of 1715ºF as the acceptance criterion for all eutectic interactions. This value is conservative with respect to NUREG 1230, which states that a eutectic reaction can occur at approximately 1736ºF, based on phase diagrams for iron-zircaloy and nickel-zircaloy.

The LBLOCA analyses model a hot pin, hot channel, and average channel. The hot pin represents a fuel pin with maximum peaking conditions. The hot channel is also modeled with the maximum peaking conditions and represents one fuel assembly minus the hot pin. The average channel is representative of the core average peaking conditions and represents the remaining assemblies in the core. Five axial peak locations along the active fuel length are typically analyzed for LBLOCA: 2.506-, 4.264-, 6.021-, 7.779-, and 9.536-ft. The peak cladding temperature (PCT) is generally related to the time it takes to quench a given location. Peak power locations higher in the core result in longer times of core uncovery. Consequently, the top of the control rods and guide tubes can be uncovered longer as well, which tends to elevate temperatures in those components. Therefore, the highest elevation for LBLOCA was

analyzed. The analyzed core power and Linear Heat Rate Limits (LHRs) also affect the analyses. A core power level of 3026 MWt is used in the generic analysis. This power level bounds the rated thermal power limit for Oconee, with significant margin. A higher initial core power increases the initial core average fuel temperatures (i.e., stored energy) and increases the decay heat generation during the transient. More stored energy and higher decay heat rates increase the steam production and hinders the core flooding rate, increasing the time of core uncovering. The LBLOCA control rod integrity analysis uses an LHR value of 17.8 kW/ft, which is the highest LHR limit for all B&W plants. Coupled with the selected core power, using the highest LHR limit maximizes the local heatup effects for the core components of interest.

During normal operations, the regulating rods (Groups 5, 6, and 7) are often partially inserted at the top of the core, with the safety rods (Groups 1, 2, 3, and 4) fully withdrawn. For the purposes of this analysis, the initial temperature of the control rods will reflect that of the rods being fully withdrawn. However, immediately at the start of the transient, the control rods will be conservatively assumed to be fully inserted to ensure that they are exposed to higher temperatures for a longer period of time. Consistent with limiting ECCS evaluation model analyses, a break at the cold-leg pump discharge (CLPD) will be the break location for both the LBLOCA and SBLOCA analyses for control rod integrity. The CLPD break location provides the worst transient results since it reduces available ECCS.

Under these conditions, the fluid temperatures and time at elevated temperature were maximized in order to minimize the control rod heat removal during the transient. The results of the LBLOCA analysis indicate that, even with conservative treatment, no control rod melt will occur for LBLOCA events with the initial conditions modeled. The maximum control rod silver-indium-cadmium absorber temperature is 1435ºF, which is less than the silver-indium-cadmium melt temperature of 1470ºF. Considering the melt temperature is not reached, the eutectic temperature is not reached (1715ºF), so even if there is contact between the control rod cladding sheath and the M5 guide tube, the control rod will remain intact. The maximum guide tube temperature attained was 1397ºF.

The generic SBLOCA survivability analyses reflect a 17.3kW/ft LHR for the hot channel, which is the highest LHR limit for SBLOCA analyses for the B&W plants. The 11-ft (10.811-ft actual) bounding axial power shape is utilized since the clad temperature is maximized when a power shape that is highly skewed to the core exit is used for SBLOCAs. These inputs and assumptions comprise a bounding set of conditions postulated on the SBLOCA analyses. The control rods, guide tubes, and flow channels are modeled using the same approach as the LBLOCA analysis.

Two power levels were analyzed for SBLOCA, 3026 MWt and 2827 MWt, to establish the sensitivity to power level. Neither case exceeded the acceptance criteria of 10 CFR 50.46, and the hot pin PCTs were 1836ºF and 1645ºF respectively. However, at 3026 MWt, the analysis predicted localized control rod absorber melting near the top of the core. This reflects the assumption of the highly skewed axial profile. Since there is a small gap between the absorber and the sheathing, there would be little relocation of any molten AIC material within the sheathing. The extent of the localized melting would be limited to only the area of direct contact and would not grossly impact the control rod or guide tube geometry. Also, the localized melt would not affect the reactivity contribution of the control rod, since it is predicted to occur very near the top of the assembly where reactivity effects are typically less pronounced with the control rods fully inserted, and a very limited volume is available to relocate melted silver-indium-cadmium material. Further, both control rod sheathing and guide tube temperatures (1640ºF and 1650ºF, respectively) remained below the eutectic temperature of 1715ºF. This indicates that the overall control rod integrity would be preserved. The generic post-LOCA control rod survivability analysis does not require any cycle specific verifications because of the

extremely bounding assumptions of core power and LHR limits used in both the LBLOCA and SBLOCA analyses.

15.14.7 Conformance with Acceptance Criteria

The NRC-approved ECCS Evaluation Models used for the LOCA analysis for Oconee class plants have been shown to be within the guidelines of 10CFR50 Appendix K. These models have been used to perform detailed sensitivity studies to assure that any adverse phenomena are identified and adequately addressed. These analyses have demonstrated that the consequences of hypothetical LOCA's up to and including a double-ended break of the largest pipe in the RCS are within the limits prescribed in 10CFR50.46, as follows:

15.14.7.1 Peak Cladding Temperature

The maximum peak cladding temperature was calculated to be 1851.9°F, which is less than the 2200°F limit.

15.14.7.2 Maximum Cladding Oxidation

The maximum local cladding oxidation was calculated to be 4.22 percent, which is less than the 17 percent limit.

15.14.7.3 Maximum Hydrogen Generation

The worst case core average hydrogen generation was calculated to be less than 0.3 percent, which is less than the 1 percent limit.

15.14.7.4 Coolable Geometry

Changes in core geometry due to thermal and irradiation effects and mechanical loading have been calculated and show that no gross core blockage or disfiguration will occur. The core will maintain a coolable geometry.

15.14.7.5 Long-Term Cooling

Subsequent to the blowdown, refill, and reflood phases of a LOCA, long-term cooling to remove core decay heat for an extended period of time must be established. The ECCS is designed to perform this function. Operator action is assumed to be available fifteen minutes following a LOCA. Several operational modes are available to provide the necessary cooling and also to assure that adequate coolant circulation exists to prevent any concentration of boric acid in a region of the RCS (Refer to Section 6.3.3.2.1). Redundancy in the design of the ECCS and multiple available flowpaths for removing core heat provide for sufficient long-term cooling.

15.14.8 Environmental Evaluation

The radiological consequences of a LOCA are bounded by the consequences of the Maximum Hypothetical Accident.

15.14.9 Conclusions

A complete spectrum of LOCAs have been conservatively analyzed with the NRC-approved evaluation models which conform to 10CFR50 Appendix K. The results of these analyses meet the acceptance criteria of 10CFR50.46. The off-site environmental consequences are within the

dose limits of 10CFR50.67. Therefore, the consequences of all design basis LOCAs have been shown to be acceptable.

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15.15 Maximum Hypothetical Accident

15.15.1 Identification of Accident

The analyses in the preceding sections have demonstrated that even in the event of a LOCA accident, no significant core melting will occur. However, to demonstrate in a still more conservative manner that the operation of a nuclear power plant at the proposed site does not present any undue hazard to the general public, a maximum hypothetical accident (MHA) involving a gross release of fission products is evaluated. No mechanism whereby such a release occurs is postulated, since this would require a multitude of failures in the engineered safeguards which are provided to prevent such an occurrence. Fission products are assumed to be released from the core as stated in Regulatory Guide 1.183 (Reference 2. The Reactor Building Spray System is credited with removal of a portion of the remaining iodine from the building atmosphere. The total core fission product inventory of interest is given in Table 15-15 (Reference 1).

15.15.2 Environmental Evaluation

The Reactor Building leak rate is assumed to be 0.20 percent per day by volume for the first 24 hours, and then 0.10 percent per day for the next 29 days. The other assumptions are consistent with Regulatory Guide 1.183 (Reference 2).

Total Effective Dose Equivalent (TEDE) doses for the 2 hour exposure at the exclusion area boundary, and for the 30-day exposure at the low population zone distance are calculated These dose consequences are within the 10 CFR 50.67 limits. A summary of the dose consequences for all transients and accidents is given in Table 15-16.

15.15.3 Effect of Washout

"HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED"

To provide a further evaluation of the suitability of the site, the effects of washout on surrounding drinking water reservoirs following the MHA are analyzed. Calculations are made for the case of continuous rain lasting 24 hr covering the general area of the reservoir and the site. The maximum washout rate as a function of distance is calculated from the following equation (Reference 3):

$$\omega_{max} = \frac{Q_o e^{-(y^2 / 2\sigma^2{}_y)}}{x e \sigma_y \sqrt{2\pi}}$$

where

ω_{max} = maximum washout rate $\frac{\text{Ci}}{\text{sec}-\text{m}^2}$

x = downwind distance (m)

σ_y = horizontal dispersion (m)

y = crosswind distance from plume axis (m)

Q_o = release rate $(\frac{Ci}{sec})$

The equation above is conservative since the results do not consider the wind speed or vertical distribution in the cloud. The wind direction is assumed to remain towards Lake Keowee for the 24 hr period with the plume center lines uniformly distributed over this section. Washout is assumed to occur under neutral stability conditions, Pasquill D, which is typical for a rainy day.

The average release rate from the Reactor Building during the 24-hr period following the accident is 0.37 equivalent curies of iodine-131 per sec. Using the above equation, the maximum iodine washout is calculated by assuming that all of the iodine that has washed out remains in the surrounding reservoir and is not affected by runoff. The average number of curies in the reservoir during a one-year period is reduced by a factor of 0.0318 due to the natural decay of iodine. Assuming that this activity mixes in the reservoir and that an adult drinks 0.8 m^3 per year (Reference 4) of the contaminated water, the total dose to the thyroid has been calculated using the methods of TID-14844. The nearest drinking water intake is approximately two miles from the site. At this distance, the total integrated one-year ingestion dose to the thyroid is 1.0 rem. This dose is well below the limits of 10CFR 100.

15.15.4 Effects of Engineered Safeguards Systems Leakage

An additional source of fission product leakage during the maximum hypothetical accident can occur from leakage of the engineering safeguards systems external to the Reactor Building during the recirculation phase for long-term core cooling. A detailed analysis of the potential leakage from these systems is presented in Section 6.1.3. A value of 12 gallons per hour (gph) leakage from LPI, HPI and BS systems was assumed in the MHA dose analysis. The MHA dose analysis also assumes back-leakage to the Borated Water Storage Tank (BWST) at a rate of 5 gallons per minute (gpm). The iodine release model in the MHA dose analysis assumes this back-leakage enters the BWST below the water level in the tank.

It is assumed that the water being recirculated from the Reactor Building sump through the external system piping contains the entire amount of iodine released from the RCS. The assumption that all of the iodine escaping from RCS is absorbed by the water in the Reactor Building is conservative since much of the iodine released from the fuel will be plated out on the building walls. It is assumed that 10 percent of all the iodine contained in the water leaking to the Auxiliary Building is released to the Auxiliary Building atmosphere.

The Auxiliary Building is ventilated and discharges to the unit vent. The activity is assumed to be continuously released from the unit vent during the recirculation phase (which is assumed not to start until 25 minutes into the event). Combined with other sources of exposure during a maximum hypothetical accident, these doses are within the guidelines specified in 10 CFR Part 50.67. Total TEDE doses from the MHA are given in Table 15-16.

15.15.5 References

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15.16 Post-Accident Hydrogen Control

15.16.1 Introduction

The purpose of this section is to summarize the analyses performed to:

Evaluate the hydrogen generation following a LOCA.

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In this section the potential for radiolytic hydrogen generation including the dose, or energy deposited in the coolant following the accident, and the basis for the selection of the hydrogen generation constant ("G" value) is analyzed. Since the FSAR analyzes the potential zircaloy-water reaction in other sections, this analysis is not presented herein and a 5 percent zirc-water reaction is assumed in the reference case described in subsequent sections. The potential for hydrogen generation from a zinc-boric acid reaction when borated water spray solution contacts galvanized steel and aluminum in the Reactor Building at the post-accident temperature is also considered. The analysis shows the radiolytic hydrogen generation rate plus the hydrogen contributed by the zircaloy and other reactions.

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Regulatory Guide 1.7 "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident" has been referenced in several sections of this analysis (Reference 14). Even though the Regulatory Guide has been used for guidance and information, Oconee is not committed to Regulatory Guide 1.7 (Reference 14).

15.16.2 Post-Accident Hydrogen Generation

Section 15.16.2 is supported by Reference 15 in its entirety.

15.16.2.1 Radiolytic Hydrogen Generation

Water radiolysis is a complex process involving reactions of numerous intermediates. However, the overall radiolytic process may be described by the reaction:

$$2H_2O \text{ ------> } 2H_2 + O_2$$

Of interest here is the quantitative definition of the rates and extent of radiolytic hydrogen production following the Design Basis LOCA. An extensive program was conducted by Westinghouse to investigate the radiolytic decomposition of the core cooling solution following the Design Basis LOCA. In the course of that investigation, it became apparent that two separate radiolytic environments exist in the Containment at Design Basis Accident conditions. In one case, radiolysis of the core cooling solution occurs as a result of the decay energy of fission products in the fuel. In the other case, the decay of dissolved fission products, which have escaped from the core, results in the radiolysis of the sump solution.

15.16.2.1.1 Core Solution Radiolysis

As the emergency core cooling solution flows through the core, it is subjected to gamma radiation by decay of fission products in the fuel. This energy deposition results in solution radiolysis, and the production of molecular hydrogen and oxygen. The initial production rate of these species will depend on the rate of energy absorption and the specific radiolytic yields.

The energy absorption rate in solution can be assessed from knowledge of the fission products contained in the core, and a detailed analysis of the dissipation of the decay energy between core materials and the solution. The results of Westinghouse studies show essentially all of the beta energy is absorbed within the fuel and cladding, and that this represents approximately 50 percent of the total beta-gamma decay energy. This study shows further that of the gamma energy, a maximum of 7.4 percent will be absorbed by the solution in the core. However, for this analysis 10 percent will be used as a conservative estimate. For the maximum credible accident case, the energy deposited in the sump accounts for the assumed TID 14844 release of 50 percent halogens and 1 percent other fission products. The noble gases are assumed by the TID 14844 model to excape to the Containment vapor space where little or no water radiolysis would result from decay of these nuclides.

For the purposes of this analysis, the calculations of hydrogen yield from core radiolysis are performed with the very conservative value of 0.45 molecules per 100 ev. This value is conservative and a maximum for this type of aqueous solution and gamma radiation is confirmed by many published works.

15.16.2.1.2 Sump Solution Radiolysis

Another potential source of hydrogen assumed for the post accident period occurs from water contained in the Containment sump being subjected to radiolytic decomposition by fission products. In this case, an assessment must be made as to the decay energy deposited in the solution and the radiolytic hydrogen yield, much in the same manner as given above for core radiolysis. The energy deposited in solution is computed using the following basis:

1. For the maximum credible accident, a TID-14844 release model is assumed where 50 percent of the total core halogens and 1 percent of all other fission products, excluding noble gases, are released from the core to the sump solution.
2. The quantity of fission products released is equal to that from a reactor operating at full power (2568 MWt) for 980 days prior to the accident.
3. The total decay energy from the released fission products, both beta and gamma, is assumed to be fully absorbed in the sump solution.

A conservative value for the hydrogen yield for sump radiolysis of 0.30 molecules per 100 ev is used in the maximum credible accident case.

15.16.2.1.3 Deleted per 2000 Update

15.16.2.2 Chemical Hydrogen Generation

In addition to the radiolytic hydrogen generation sources (core and sump radiolysis) following a Design Basis Accident, hydrogen may also be evolved from two chemical sources: (1) zirconium-water reaction involving clad material, and (2) from the reaction of zinc and aluminum within the Reactor Building with the borated coolant water.

15.16.2.2.1 Method of Analysis

The quantity of zirconium which reacts with the core cooling solution depends on the performance of the Emergency Core Cooling System. 10CFR50.46(b)(3) states that the total

amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 1% of the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

Aluminum is more reactive with the Reactor Building spray solution than other plant materials such as galvanized steel, copper, and copper-nickel alloys. However, because of the relatively large amount of exposed galvanized and zinc-based painted surfaces in the Reactor Building, zinc corrosion must be considered as a contributing hydrogen source.

It should be noted that zirconium-water reaction and the aluminum and zinc corrosion with Reactor Building spray are chemical reactions and thus essentially independent of the radiation field inside the Reactor Building following a LOCA. Radiolytic decomposition of water is dependent on the radiation field intensity. The radiation field inside the Reactor Building is calculated for the maximum credible accident in which the fission product activities given in TID-14844 are used.

15.16.2.2.2 Typical Assumptions

The following discussion outlines the assumptions used in the calculations.

15.16.2.2.3 Zirconium-water Reaction

Hydrogen can be generated during a LOCA by the reaction of hot zirconium cladding with the surrounding steam. The zirconium-water reaction is described by the chemical equation:

$Zr + 2H_2O$ ------> $ZrO_2 + 2H_2$ + Heat

The quantity of zirconium which reacts with the core cooling solution depends on the performance of the Emergency Core Cooling System (ECCS). For Oconee the maximum of 1% zirconium-water reaction is assumed. Regulatory Guide 1.7 requires that the assumption for hydrogen produced from the zirconium-water reaction equal 5 times the extent of the maximum calculated reaction under 10CFR50.46, i.e., 5.0%. Per Regulatory Guide 1.7, the zirconium-water hydrogen source is assumed to be released over a 2 minute period from the start of the transient, and is assumed to be distributed uniformly throughout Containment.

15.16.2.2.3.1 Corrosion of Plant Materials

Another possible source of hydrogen could occur from metal surfaces exposed to an environment containing high-temperature steam, corrosive sprays, fission products, and radioactivity. Such exposure might result in surface corrosion reactions that produce hydrogen. Corrosive tests have been performed to determine the behavior of various metals that are used in Containment when exposed to a post-LOCA environment. As applied to the quantitative definition of hydrogen production rates, the results of the corrosion tests have shown that only aluminum will corrode at a rate that will significantly add to the hydrogen accumulation in the Containment atmosphere. However, because of the relatively large amount of exposed galvanized and zinc-based painted surfaces in Containment, zinc corrosion must be considered as a contributing hydrogen source.

The corrosion of aluminum and zinc may be described by the following reactions:

$2Al + 3H_2O$ -----> $Al_2O_3 + 3H_2$

$Zn + H_20$ -----> $ZnO + H_2$

The time-temperature cycle considered in the calculation of aluminum and zinc corrosion are based on a conservative representation of the postulated post accident Containment transient. The corrosion data points include the effects of temperature, alloy, and spray solution conditions. NOTE: In Section 5, Part C of Regulatory Guide 1.7 it is stated that values given in Table 1 for evaluating production of combustible gases following a LOCA may be changed on the basis of additional experimental evidence and analyses. As a result the minimum assumed value give for aluminum corrosion rate of 200 mpy is not used in the analysis.

15.16.2.3 Primary Coolant Hydrogen

The quantity of hydrogen assumed in the primary coolant is 450 scf. This value is expected to bound the total of the hydrogen dissolved in the coolant water at and corresponding equilibrium hydrogen in the pressurizer gas space. The 450 scf of hydrogen is assumed to be released immediately into Containment at the initiation of the LOCA.

15.16.3 EVALUATION OF HYDROGEN CONCENTRATIONS

15.16.3.1 Hydrogen Flammability Limits

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The hydrogen generation which occurs following a design basis LOCA is a slow process driven by sump radiolysis and metal corrosion (Reference 15). The concentration thirty days following a design basis LOCA is approximately 6.4 volume percent. Studies of containment structural capacity and the effects of hydrogen combustion have shown concentrations much higher than 4 volume percent are required to threaten the integrity of a large dry containment like the Oconee containments. Furthermore, studies have shown that the majority of risk to the public is from accident sequences that lead to containment failure or bypass, and that the contribution to the risk from accident sequences involving hydrogen combustion is actually quite small for large, dry containments such as Oconee's. This is true despite the fact that hydrogen produced in these events is substantially larger than the hydrogen production postulated by 10 CFR 50.44(d) and RG 1.7 (Reference 26). NUREG/CR-4551 also states that hydrogen combustion in the period before vessel failure is now generally considered to present no threat to large, dry containments.

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15.16.3.2 Evaluation of Hydrogen Concentrations

Prediction of hydrogen generation following the loss-of-coolant accident using the assumptions and method of analysis described in Section 15.16.2 shows that although hydrogen production rate decreases as the post-accident time increases, total hydrogen accumulation can exceed the lower flammability limit of 4 volume percent. The analysis shows that using conservative assumptions, post-LOCA hydrogen concentrations can reach 3 volume percent in approximately 216 hours (9 days) and 4 volume percent in approximately 360 hours (15 days) (Reference 15).

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Post accident hydrogen concentrations are indicated by the Containment Hydrogen Monitoring System (CHMS). The CHMS is described in Section 9.3.7 and is shown in Figure 9-15. This instrumentation provides two redundant channels of hydrogen monitoring that can monitor hydrogen concentrations at different levels of the containment including CHRS inlet and return concentrations.

In order to assure high concentration pockets of hydrogen do not exist and that representative samples of hydrogen can be obtained, adequate mixing of hydrogen throughout containment should exist. Mixing in the Reactor Building atmosphere is expected to be good. The Reactor Building cooling fans or sprays will introduce considerable turbulence to the building atmosphere to provide good mixing of hydrogen in the early stages of the accident. In addition, all the Reactor Building volumes are connected by large vent areas (stair wells, elevator shafts, grating) to promote good air circulation.

Figure 15-89 shows the Reactor Building cross-section. The hydrogen generated will be primarily from the corrosion of metals in the large open area of the containment and from radiolysis of water in the sump and water leaking from the RCS. These locations are within the unrestricted main volume of the building and will permit the hydrogen to diffuse rapidly and provide a uniform mixture in this area. This rapid mixing occurs because hydrogen has a high diffusion rate and a low generation rate, and is capable of diffusing in all directions. The hydrogen will diffuse very rapidly giving an even distribution under the conditions existing in the Reactor Building. This situation is not analogous to one where attempts are made to mix streams of gases under dynamic conditions where residence times and mixing distances are critical. In addition, the thermal mixing effects, heating of air above the hot sump water, and possible steam releases from the RCS will move the hydrogen laden air from the points of generation toward the cold external walls and emergency cooling equipment. Although hydrogen is lighter than air, it will not tend to concentrate in high areas because of the high diffusion rate and because of the open design of the Reactor Building.

Since the hydrogen is generated primarily from corrosion of metals and core radiolysis in the large open areas, the hydrogen must diffuse from the major volumes into those minor volumes which are enclosed. The minor volumes or those not having good communication with the major volumes would be at a lower hydrogen concentration because the hydrogen is diffusing from the higher concentration level to a lower concentration level. Accordingly, pockets, if they exist, will be low concentration pockets rather than high concentration pockets.

The ability of hydrogen to diffuse rapidly into all volumes is inferred by a condensing steam environment (CSE) experiment (Reference 8) which measured the spatial concentration of iodine in the various compartments. The tests showed very good mixing in the main chamber and a rapid interchange by diffusion and mixing with the atmosphere of other chambers which had limited communication. The diffusivity of hydrogen is approximately 10 times that of iodine so a more uniform mixture would be expected for hydrogen than for iodine. Also, the higher concentrations would provide greater concentration gradients for better diffusion than was indicated by the CSE tests.

During a DBA LOCA, the operation of Reactor Building sprays and RBCUs will provide mixing in containment. This along with the fact that the hydrogen generation rates are low for the majority of the accident support the conclusion that a nearly uniform hydrogen concentration will exist in containment.

Hydrogen concentrations on the order of 6 percent or less are bounded by hydrogen generated during a severe accident and would not be a threat to containment integrity since there is ample time between burns to reduce elevated containment temperatures using the installed containment heat removal systems. Based on analysis, Oconee could withstand the consequences of uncontrolled hydrogen-oxygen recombination without loss of safety function with up to 100 percent metal-water reaction.

15.16.4 Deleted per 2003 update

15.16.5 Deleted per 2003 update

15.16.6 Conclusions

Figure 15-175 shows that if no measures were taken to control hydrogen accumulation in the Reactor Building, the hydrogen concentration within the Reactor Building can be expected to reach the lower flammability limit of 4 volume percent at approximately 360 hours (Reference 15).

Based on analysis, Oconee could withstand the consequences of uncontrolled hydrogen-oxygen recombination without loss of safety function with up to 100 percent metal-water reaction.

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15.17 Small Steam Line Break Accident

15.17.1 Identification of Causes and Description

The small steam line break accident is caused by small breaks in the steam lines or by failures of valves connected to the steam lines. The break flowrate, the reactor kinetic behavior, and the status of the control systems have a large effect on the plant response. The initial plant response to the increase in steam flow is a decrease in steam generator pressure and an overcooling of the Reactor Coolant System (RCS). The expected plant response with the Integrated Control System (ICS) in automatic would be for the main turbine control valves to close to return turbine header pressure to the setpoint, the control rods would insert to offset the increase in the reactor power due to the negative moderator coefficient of reactivity, and main feedwater (MFW) flow would be controlled to maintain the secondary heat sink in balance with the reactor power. This automatic response may be successful in not tripping the reactor. With the ICS in automatic or manual control, a reactor trip on high neutron flux, flux/flow/imbalance, variable low pressure-temperature, on turbine trip due to main feedwater pump trip, or by manual operator action would be expected.

The small steam line break accident analyses assume that the ICS is in manual control for initial conditions of full power with four reactor coolant pumps (RCPs) in operation, and 75% power with three RCPs in operation. The ICS in manual control is more limiting than with the ICS in automatic. A range of break sizes and moderator temperature coefficients are analyzed to determine the combination that approaches the most limiting conditions relative to the DNBR limit. The effect of a decrease in the reactor vessel downcomer temperature on the indicated excore power range flux is modeled. Several non-safety systems could cause a trip of the MFW pumps thereby mitigating the consequences of the transient. These include AFIS circuitry (which actuates some non-safety grade components), the ICS high steam generator level trip, and the low MFW pump discharge pressure trip. None of these non-safety systems are credited in the analyses. The results presented model the replacement steam generators. The analysis methodology and the computer codes used in this analysis are given in Table 15-33. The initial conditions are given in Table 15-34. The Reactor Protective System and Engineered Safeguards Protective System setpoints and delay times are given in Table 15-35. It is conservatively assumed that offsite power is maintained since a loss of offsite power would result in reactor trip. Operator action is credited with manually tripping the reactor at 10 minutes if an automatic reactor trip has not occurred. No single failure has been identified which adversely affects this transient.

A small steam line break accident is considered to be either a fault of moderate frequency (valves failing open) or an infrequent fault (pipe break). To bound both types of events, the analysis assumes pipe breaks as initiating events, with acceptance criteria corresponding to the less severe fault of moderate frequency category. The acceptance criteria for this accident are that the minimum DNBR remains above the limit (1.34 for four and three RCP operation for the Mk-B-HTP fuel type), that the centerline fuel melt limit is not exceeded, and that the offsite doses will be within 10% of the 10CFR50.67 limits.

15.17.2 Analysis

The RETRAN system thermal-hydraulic analysis results are valid for the full core with Mk-B-HTP fuel. The limiting small steam line break accident for DNB considerations is a break size of 1.46 ft^2 initiated from four RCP operation, with a moderator temperature coefficient of -7 pcm/°F. The transient response is given in Figure 15-168, Figure 15-169, Figure 15-170, Figure 15-171,

Figure 15-172 and Figure 15-173 and the sequence of events is given in Table 15-49. The duration of the analysis is 600 seconds, which includes the core conditions of minimum DNBR margin. The blowdown out the break increases the steam flow exiting the steam generators by approximately 27% (Figure 15-168). The steam generator pressure decrease (Figure 15-169) propagates throughout the secondary system, causing main feedwater flow to increase (Figure 15-170) and a decrease in main feedwater temperature. RCS temperatures decrease (Figure 15-171) causing a power increase (Figure 15-172) due to the negative moderator temperature coefficient of reactivity. The moderator and Doppler feedback mitigates the power excursion. The transient reaches a sustained power level of approximately 139%. The high flux and the flux/flow/imbalance trips do not actuate due to the effect of the decrease in the reactor vessel downcomer temperature. The RCS pressure response (Figure 15-173) follows RCS average temperature. RCS pressure eventually increases due to backup heaters energizing and increased makeup flow. The system analysis results are input to a detailed core thermal-hydraulic analysis assuming a standard reference power distribution. The transient minimum DNBR is 1.435, which is greater than the design limit, for the full core with Mk-B-HTP fuel. A fuel pin census analysis is performed to determine and affirm if DNBR margin exists or the number of fuel pins that exceed the DNBR limit. The results of the fuel pin census analysis for the small steam line break accident is that DNB margin exists for all of the fuel pins. Thus, no fuel failure is expected. The centerline fuel melt limit has been evaluated and it is not violated.

15.17.3 Environmental Consequences for the Small Steam Line Break

A conservative consequences analysis is performed for a postulated break of a small steam line or an auxiliary steam line. This break results in an increased thermal demand on the reactor coolant system (RCS) and a rapid cooldown and positive reactivity addition from a negative temperature coefficient. This transient is not postulated to induce fuel failures, steam generator tube failures or any other failures of fission product barriers or primary system pressure boundaries, or any other pieces of equipment. Thus, the environmental consequences result from plant releases of pre-existing RCS activity transported to the secondary side by postulated steam generator tube leakage, and of pre-existing secondary activity. This activity is then released to the environment by releases associated with the normal operation of plant equipment or the operation of plant equipment as intended in response to the accident, and as part of the subsequent cooldown activities.

Two RCS source terms are examined as part of this analysis: a preaccident iodine spike and a concurrent iodine spike. The first models the maximum Dose Equivalent Iodine (DEI) activity concentration permitted by Technical Specifications for an iodine spike at full power. This preaccident spike is postulated to occur at the time of accident initiation. This source term is modeled to be released instantaneously and homogeneously such that the RCS activity is in equilibrium at the start of the accident. The second source term models a concurrent iodine spike, where the primary system transient associated with the accident causes an iodine spike in the primary system. The increase in primary coolant iodine concentration uses a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value specified in Technical Specifications. Both iodine spike source terms also bound Technical Specification limits for Dose Equivalent Xenon (DEX).

An initial source term is also modeled for the secondary side. The maximum Technical Specification allowed DEI concentration is modeled to be present in the secondary side water, the steam generators and any makeup water supplied to the unit. Thus, the secondary side is essentially modeled as an infinite source of water at the secondary side Technical Specification DEI concentration limit.

In order to transport and release primary activity to the environment, a primary to secondary release path is modeled in the steam generators. This path is postulated to exist at the start of the accident, but is not caused by the steam line break. The tube leakage into the unaffected steam generator modeled bounds the maximum allowed tube leakage rate into one steam generator. The affected steam generator is modeled with a leakage rate that bounds the maximum allowed unidentified primary to secondary leakage allowed by Technical Specifications.

The thermal/hydraulic model discussed in the previous sections is used as the basis for the plant response and steam releases modeled in the environmental analysis. The plant is initially operating in a normal mode at full power (plus maximum thermal power uncertainty) with primary to secondary leakage. The only releases occurring at the start of the accident are from the condensate steam air ejectors (CSAEs), which discharge a mixture of motive steam and condensate gases. Since the CSAEs operate continuously, no gases are assumed to be in the secondary system, as they would be removed by the CSAEs when introduced into the secondary system. When the break initiates, the activities in the primary and secondary side are modeled to be instantaneously and homogeneously released to their respective systems. The reactor is manually tripped by the operators after allowing for the maximum postulated time for them to identify the accident. Radioactive decay (and daughter product production) is then begun in the model. The affected steam generator begins to discharge all of its activity directly to the environment. The unaffected steam generator also discharges its inventory directly to the environment through the break until the Turbine Stop Valves close shortly after reactor trip. This steam header will repressurize resulting in lifting its Main Steam Relief Valves. Since the steam release from the affected steam generator is not isolable, this release will continue as long as water and conditions conducive to boiling exist in this steam generator.

In order to maximize releases to the environment, the condenser is assumed to not be available. This requires that the unit be cooled down using the unaffected steam generator by discharging steam from this steam generator directly to the environment through the Atmospheric Dump Valves (ADVs). No credit is taken for the condenser and no partitioning credit is taken for CSAE releases which are modeled to occur until the beginning of cooldown.

The small steam line break does not cause the Turbine Driven Emergency Feedwater Pump (TDEFWP) to start. Thus, there is no discharge to the environment from the TDEFWP exhaust, and therefore, this release path is not included in the environmental analysis.

Since Oconee Nuclear Station is a B&W designed plant, it uses once through steam generators which provide for vertical tubing which carries primary coolant from the top of the generator to its bottom while exchanging heat with the secondary fluid on the shell side. Because of this tubing arrangement, the tube leakage is modeled to occur above the secondary water mass in the steam generator. Therefore, no credit is taken for iodine partitioning in the steam generator. No credit is taken for iodine plateout in the steam lines or any other surface.

The thermal/hydraulic response of the plant to a small steam line break does not result in the need for a soak prior to cooldown. Thus, after the plant is stabilized, cooldown can be commenced at the maximum rate permitted by Technical Specifications. This rate is reduced as required by Technical Specifications at the appropriate temperature. When the thermodynamic conditions are met for the Low Pressure Injection (LPI) system to remove decay heat from the primary, cooldown releases from the ADVs cease and decay heat removal is accomplished by the LPI system. Primary to secondary leakage and its release to the atmosphere continue until the temperature of the primary water leaking is less than the boiling point for water at atmospheric conditions. At this point all releases of activity from the plant model cease.

Offsite atmospheric dispersion factors from the Updated Final Safety Analysis Report Chapter 2 were used. Dose conversion factors from Federal Guidance Reports 11 and 12 were used.

Based upon this model, releases of activity to the environment from the primary and secondary systems can be calculated and used to calculate doses at the Exclusion Area Boundary (EAB), the Low Population Zone (LPZ), and in the Control Room. The doses calculated meet the regulatory criteria of 10 CFR 50.67 for each of the source terms examined. The results are presented in Table 15-16.

15.17.4 Conclusions

The small steam line break accident analysis results show that DNBR margin exists for all of the fuel rods, and that no fuel failures due to centerline fuel melt occur. The environmental consequences meet the acceptance criteria. All of the acceptance criteria are met.

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15.18 Anticipated Transients Without Trip

An anticipated transient without trip (ATWT) or anticipated transient without scram (ATWS) is an anticipated operational occurrence (such as loss of feedwater, loss of condenser vacuum, or loss of offsite power) that is accompanied by a failure of the reactor trip system to shutdown the reactor. Studies on ATWS at B&W plants showed that an alternate method is required to provide a scram and initiate turbine trips and auxiliary feedwater flow.

The effects of ATWS are not considered as part of the design basis for transients analyzed in Chapter 15.0. The final USNRC ATWS rule requires that all US B&W - designed plants install a diverse scram system (DSS) to initiate control rod insertion, and ATWS mitigation system actuation circuitry (AMSAC) to initiate a turbine trip and actuate auxiliary feedwater, independent of the reactor trip system. The AMSAC and DSS are part of the ATWS Mitigation System described in Section 7.8.

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