CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9906030238 DOC.DATE: 99/05/27 NOTARIZED: YES DOCKET #
FACIL:50-269 Oconee Nuclear Station, Unit 1, Duke Power Co. 05000269
50-270 Oconee Nuclear Station, Unit 2, Duke Power Co. 05000270
50-287 Oconee Nuclear Station, Unit 3, Duke Power Co. 05000287

AUTH.NAME AUTHOR AFFILIATION
MCCOLLUM,W.R. Duke Power Co.
RECIP.NAME RECIPIENT AFFILIATION
Records Management Branch (Document Control Desk)

SUBJECT: Provides markups of UFSAR to document results of dose analyses & revises NSHC & EIS provided in 990405 submittal to satisfy first commitment.

DISTRIBUTION CODE: A001D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 48
TITLE: OR Submittal: General Distribution

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	LPD2-1 LA	1	1	LPD2-1 PD	1	1
	LABARGE,D	1	1			
INTERNAL:	ACRS	1	1	FILE CENTER 01	1	1
	NRR/DE/EEIB	1	1	NRR/DE/EMCB	1	1
	NRR/DE/EMEB	1	1	NRR/DSSA/SPLB	1	1
	NRR/DSSA/SRXB	1	1	NRR/SPSB JUNG,I	1	1
	NUDOCS-ABSTRACT	1	1	OGC/RP	1	0
EXTERNAL:	NOAC	1	1	NRC PDR	1	1

CATEGORY 1 DOCUMENT

MICROFILMED

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE. TO HAVE YOUR NAME OR ORGANIZATION REMOVED FROM DISTRIBUTION LISTS OR REDUCE THE NUMBER OF COPIES RECEIVED BY YOU OR YOUR ORGANIZATION, CONTACT THE DOCUMENT CONTROL DESK (DCD) ON EXTENSION 415-2083

TOTAL NUMBER OF COPIES REQUIRED: LTTR 15 ENCL 14

Duke Energy

W. R. McCollum, Jr.
Vice President

Duke Energy Corporation
Oconee Nuclear Station
7800 Rochester Highway
Seneca, SC 29672
(864) 885-3107 OFFICE
(864) 885-3564 FAX

May 27, 1999

U. S. Nuclear Regulatory Commission
Washington D. C. 20555-0001

ATTENTION: Document Control Desk

Subject: Duke Energy Corporation
Oconee Nuclear Station, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Supplement to Unit 2 Cycle 18 Reload Report and
License Amendment Request (TSC 99-06)

On April 5, 1999, Duke Energy Corporation (Duke) submitted a license amendment request (LAR) for Facility Operating License Nos. DPR-38, DPR-47, and DPR-55 for Oconee Nuclear Station Units 1, 2, and 3, respectively, pursuant to 10 CFR 50.90. The LAR proposed changes to the Technical Specifications, Bases, Updated Final Safety Analysis Report (UFSAR), and Core Operating Limits Report (COLR) to incorporate Topical Report DPC-NE-3005-P, "Thermal-Hydraulic Transient Analysis Methodology." It supports the future operation of Oconee Unit 2 during Cycle 18 (O2C18).

In the LAR, Duke made the following commitments:

1. The dose methodology analysis and associated UFSAR marked pages will be submitted to the NRC by May 31, 1999; and

2. The LAR that includes relocating UFSAR Section 15.13.4 to UFSAR Chapter 5 will be submitted to the NRC by April 30, 1999.

This submittal satisfies the first commitment discussed above. The dose analyses associated with the new and revised accident analyses to support operation of Oconee Unit 2 during Cycle 18 have been completed. Accordingly, this submittal provides the markups of the Updated Final Safety Analysis Report (UFSAR) to document the results of

A001

the dose analyses. Also, this submittal revises the No Significant Hazards Consideration and the Environmental Impact Statement provided in the April 5, 1999, submittal to reflect the results of the dose analyses.

On April 26, 1999, Duke satisfied commitment No. 2 by submitting an independent LAR that included a proposal to relocate UFSAR Section 15.13.4 to Chapter 5.

One item from the original April 5, 1999, LAR remains to be completed. The analyses to justify a 5°F ΔTc are still ongoing. Upon completion of these analyses, Duke will supplement the LAR by revising the applicable limits. Duke commits to submit this information by September 1, 1999.

This submittal contains the following attachments:

Attachment 1 provides a marked copy of the affected UFSAR pages. Permanent changes to the UFSAR will be made in accordance with 10CFR50.71(e).

Attachment 2 provides a description of and technical justification for the proposed changes.

Attachment 3 provides a revision to the No Significant Hazards Consideration provided in the LAR dated April 5, 1999.

Attachment 4 provides a revision to the categorical exclusion from performing an Environmental Assessment/Impact Statement provided in the LAR dated April 5, 1999.

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, the changes contained in this LAR have been reviewed and approved by the Oconee Plant Operations Review Committee and the Duke Corporate Nuclear Safety Review Board.

Pursuant to 10CFR50.91, a copy of this supplement is being sent to the State of South Carolina.

Inquiries on this matter should be directed to J. S. Warren at (704) 382-4986.

Very truly yours,

W. R. McCollum, Jr., Site Vice President
Oconee Nuclear Site

Attachments

xc w/attachments:

L. A. Reyes, Regional Administrator
U. S. Nuclear Regulatory Commission, Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

D. E. Labarge, NRC Senior Project Manager (ONS)
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, DC 20555-0001

M. A. Scott
NRC Senior Resident Inspector (ONS)

V. R. Autry, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
South Carolina Department of Health and
Environmental Control
2600 Bull Street
Columbia, SC 29201

AFFIDAVIT

W. R. McCollum, being duly sworn, states that he is Site Vice President of Duke Energy Corporation; that he is authorized on the part of said Corporation to sign and file with the Nuclear Regulatory Commission revisions to the Oconee Nuclear Station Facility Operating Licenses No. DPR-38, DPR-47, and DPR-55; and that all the statements and matters set forth herein are true and correct to the best of his knowledge.

W. R. McCollum, Jr., Site Vice President

Subscribed and sworn to before me this 27 day of

May, 1999

Notary Public

My Commission Expires:

2-12-2003

SEAL

ATTACHMENT 1

MARKUP OF THE UPDATED FINAL SAFETY ANALYSIS REPORT

Insert 1

(cont.)

Insert 4

Insert H

INSERT H

- 15-50 Iodine and Noble Gas Inventory in Reactor Core and Fuel Rod Gaps – 500 EFPD Equilibrium Cycle
- 15-51 Reactor Coolant System Fission Product Source Activities – 500 EFPD Equilibrium Cycle
- 15-52 Parameters for Steam Generator Tube Rupture Accident Dose Analysis
- 15-53 Parameters for Postulated Rod Ejection Offsite Dose Analysis
- 15-54 Parameters for Large Main Steam Line Break Accident Dose Analysis
- 15-55 Parameters for Small Main Steam Line Break Accident Dose Analysis

Table 2-29. Dispersion Factors Used for Accident and Routine Operational Analyses X/Q (second m^{-3})

At Exclusion Area Boundary 1 - mile (1609 M)			
	0-2 Hours	0-24 Hours	0-7 Days
Ground Releases	1.16×10^{-4}		
Elevated Releases	3.35×10^{-5}	9.73×10^{-6}	2.98×10^{-6}
At Boundary of Low Population Zone			
	0-24 Hours	0-30 Days	
Ground Releases*	1.32×10^{-5}	7.2×10^{-7}	
Elevated Releases**	3.90×10^{-6}	3.42×10^{-7}	
Elevated Releases**	4.15×10^{-6}	3.70×10^{-7}	
Long-Term (One Year) Exclusion Area Boundary			
Ground Release	4.61×10^{-6}		
Elevated Release	8.74×10^{-7}		

Note:

*At valley construction 6 1/2 miles (10,464 m) from site near Boundary of Low Population Zone.

** Six miles (9,658 m) from site at Boundary of Low Population Zone.

Table 2-29. Dispersion Factors Used for Accident and Routine Operational Analyses χ/Q

At Exclusion Area Boundary (1609 m)

	0-2 hrs	0-24 hrs	0-7 days
Ground Release	1.16 E-4		
Elevated Release	3.35 E-5	9.73 E-6	2.98 E-6

At Boundary of Low Population Zone (9650 m)

	0-24 hrs.	0-30 days
Ground Releases*	1.32 E-5	7.2E-7
Elevated Releases**	3.90 E-6	3.42 E-7

Long-Term (One Year) Exclusion Area Boundary

Ground Release	4.61 E-6
Elevated Release	8.74 E-7

The Following Are Applicable to the Unit 2 Chapter 15 Accident Analyses

At Exclusion Area Boundary (1609 m)

	0-2 hrs
Ground Releases	2.20E-4
Elevated Releases	3.35E-5

At Boundary of Low Population Zone (9650 m)

	0-8 h	8-24 h	1 d - 4 d	4 d - 30 d
Ground Releases	2.35E-5	4.70E-6	1.50E-6	3.30E-7
Elevated Releases	N/A	N/A	N/A	N/A

* at valley construction 10,464 m from site near Boundary of LPZ

** 9,658 m from site at Boundary of LPZ

Table 2-30. Dispersion Factors. Of Table 2-29 to be Used During Various Release Conditions

	Release Condition	Appropriate Dispersion Factor
(a)	Fuel handling accident	0-2 hour elevated release at exclusion area boundary
(b)	Steam Line Failure	0-2 hour ground release at exclusion area boundary
(c)	Rod Ejection Accident	0-2 hours at site boundary; 0-24 hours and 0-30 days at boundary of low population zone
(d)	Loss-of-Coolant Accident (assume 50 percent ground release and 50 percent elevated release after 90 percent iodine removal by filtration)	Same as (c)
(e)	Maximum Hypothetical Accident (MHA)	Same as (c)
(f)	Loss of Electric Power	Same as (a)
(g)	Loss-of-Coolant Accident during purge	Same as (a)
(h)	Engineered Safeguards Leakage	Same as (a)
(i)	Lifetime Shim Bleed (continuous release)	Long-term elevated releases at exclusion area boundary
(j)	Start-up expansion (7-day release)	0-7 day elevated releases at exclusion area boundary
(k)	Reactor Building Purge	0-24 hour elevated release at exclusion area boundary
(l)	Steam Generator Tube Failure	Same as (a)
(m)	Steam Generator Tube Leakage	Long-term elevated releases at exclusion area boundary
(n)	Pressurizer and Letdown Storage Tank Venting	0-7 day elevated release at exclusion area boundary.
(o)	Waste - Gas - Tank Rupture	0-2 hour elevated release at exclusion area boundary.

Table 2-30. Dispersion Factors of Table 2-29 to be used During Various Release Conditions		
	Release Condition	**Appropriate Dispersion Factor**
(a)	Fuel Handling Accident	0-2 hour elevated release at exclusion area boundary
(b)	Steam Line Failures	0-2 hour ground release at exclusion area boundary for steam line releases 0-2 hour elevated release at exclusion area boundary for unit vent releases 0-8 hours, 8-24 hours, 1-4 days, and 4-30 days at boundary of low population zone
(c)	Rod Ejection Accident	0-2 hour ground release at exclusion area boundary for steam line releases 0-2 hour elevated release at exclusion area boundary for unit vent releases 0-8 hours, 8-24 hours, 1-4 days, and 4-30 days at boundary of low population zone
(d)	Loss-of-Coolant Accident (assume 50 percent ground release and 50 percent elevated release after 90 percent iodine removal by filtration)	0-2 hour ground release at exclusion area boundary for steam line releases 0-2 hour elevated release at exclusion area boundary for unit vent releases 0-8 hours, 8-24 hours, 1-4 days, and 4-30 days at boundary of low population zone
(e)	Maximum Hypothetical Accident (MHA)	0-2 hour ground release at exclusion area boundary for steam line releases 0-2 hour elevated release at exclusion area boundary for unit vent releases 0-8 hours, 8-24 hours, 1-4 days, and 4-30 days at boundary of low population zone
(f)	Engineered Safeguards Leakage	0-2 hour elevated release at exclusion area boundary
(g)	Lifetime Shim Bleed (continuous release)	Long-term elevated releases at exclusion area boundary
(h)	Start-up Expansion (7-day release)	0-7 day elevated release at exclusion area boundary
(i)	Reactor Building Purge	0-24 hour elevated release at exclusion area boundary
(j)	Steam Generator Tube Failure	0-2 hour ground release at exclusion area boundary for steam line releases 0-2 hour elevated release at exclusion area boundary for unit vent releases 0-8 hours, 8-24 hours, 1-4 days, and 4-30 days at boundary of low population zone
(k)	Steam Generator Tube Leakage	Long-term elevated releases at exclusion area boundary
(l)	Pressurizer and letdown Storage Tank Venting	0-7 day elevated release at exclusion area boundary
(m)	Waste Gas Tank Rupture	0-2 hour elevated release at exclusion area boundary

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) The turbine trip circuitry has two channels, one with a one second response time, and one with a fifteen second response time. The faster response time is credited in the methodology. The turbine trip circuitry is not completely safety-grade.

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

~~(TO BE SUBMITTED LATER)~~ INSERT A

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 References

1. UFSAR Chapter 15 Transient Analysis Methodology, DPC-NE-3005-P, Revision 1, Duke Power, February 1999
2. BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants, BAW-10192P, B&W Nuclear Technologies, February 1994
3. B&W's Small-Break LOCA ECCS Evaluation Model, BAW-10154P, Babcock & Wilcox, November 1982
4. Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000-PA, Revision 1, Duke Power, December 1997

Insert A

15.1.10 Environmental Consequences Calculation Methodology

Environmental Consequences

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

Fission Product Inventories

Inventory in the Core: End-of-cycle fission product inventories within the core are calculated by the ORIGEN-2 Code (Section 15.1, Ref. 27) using a data library for extended burnup cores (Section 15.1, Ref. 28), at the instant before shutdown. The core inventories are shown in Table 15-50.

Inventory in the Fuel Pellet Clad Gap: The fuel pin gap activities were determined using Regulatory Guide 1.25 (Section 15.1, Ref. 35) and Regulatory Guide 1.77 (Section 15.1, Ref. 36). Specifically, 10 percent of the iodine activities, 30 percent of the Kr-85 activity, and 10 percent of the remaining noble gas activities in the core are accumulated in the fuel-clad gap. The environmental consequences of the loss of coolant accident, 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 core and associated gap inventories used for the LOCA, control rod cluster assembly ejection, and fuel handling accidents are shown in Table 15-50.

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, Ref. 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.

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.

Insert A Continued

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.

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).

5. RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 4, EPR, November 1988
6. VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM, Revision 3, EPRI, August 1989
7. CASMO-3: A Fuel Assembly Burnup Program User's Manual, NFA-88/48, Studsvik of America, September 1988
8. SIMULATE-3: Advanced Three-Dimensional Two-Group Reactor Analysis Code, SOA-92/01, Studsvik of America, April 1992
9. SIMULATE-3 Kinetics Theory and Model Description, SOA-96/26, Studsvik of America, April 1996
10. TACO-3 - Fuel Pin Thermal Analysis Code, BAW-10162-PA, Babcock & Wilcox, November 1989
11 Nuclear Design Methodology Using CASMO-3 / SIMULATE-3P, DPC-NE-1004-A, Duke Power, November 1992
12. Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2003-PA, Duke Power, October 1989
13. Thermal-Hydraulic Statistical Core Design Methodology, DPC-NE-2005-PA, Revision 1, Duke Power, November 1996
14. Appendix K to Part 50 - ECCS Evaluation Models, Code of Federal Regulations, Volume 10
15. RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis, BAW-10164P, Revision 3, B&W Nuclear Technologies, July 1996
16. CONTEMPT - Computer Program for Predicting Containment Pressure, BAW-10095A, Rev. 1, Babcock & Wilcox, April 1978
17. REFLOD3B - Model for Multinode Core Reflooding Analysis, BAW-10171P, Revision 2, Babcock & Wilcox, January 1989
18. BEACH -A Computer Code for Reflood Heat Transfer During LOCA, BAW-10166P, Rev. 2, Babcock & Wilcox, September 1989
19. CRAFT2 - FORTRAN Program for Digital Simulation of a Multinode Reactor Plant During Loss of Coolant, BAW-10092PA, Revision 3, Babcock & Wilcox, July 1985
20. FOAM2 - Computer Program to Calculate Core Swell Level and Mass Flow Rate During Small-Break LOCA, BAW-10155A, Babcock & Wilcox, October 1987
21. THETA1-B - A Computer Code for Nuclear Reactor Core Analysis, BAW-10094, Babcock & Wilcox, July 1974
22. Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors, 10 Code of Federal Regulations, Part 50.46
23. BWC Correlation of Critical Heat Flux, BAW-10143-PA, Babcock & Wilcox, April 1985
24. The BWU Critical Heat Flux Correlation, BAW-10199-PA, April 1996
25. American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1979, American Nuclear Society, August 1979
26. ARROTTA: Advanced Rapid Reactor Operational Transient Analysis, EPRI NP-7375-CCML, Rev. 1, EPRI, August 1993.

INSERT B

Insert B

References

27. A. G. Croff, "A Users' Manual for the ORIGEN2 Computer Code," ORNL/TM-7175 (CCC-371), July 1980.

28. S. B. Ludwig, J. P. Renier, "Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code," ORNL/TM-11018, December 1989.

29. Frank, P.W., et al., Radiochemistry of Third PWR Fuel material Test - X-1 Loop NRX Reactor, WAPD-TM-29, February, 1957.

30. Eichenberg, J.D., et al, Effects of Irradiation on Bulk UO2, WAPD-183, October, 1957.

31. Allison, G.M., and Robertson, R.F.S., The Behavior of Fission Products in Pressurized-Water Systems. A Review of Defect Tests on UO2 Fuel Elements at Chalk River, AECL-1338, 1961.

32. Allison, G.M., and Roe, H.K., The Release of Fission Gases & Iodines from Defected UO2 Fuel Elements of Different Lengths, AECL-2206, June, 1965.

33. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR 50, Appendix I," Rev. 1, October 1977.

34. The Code of Federal Regulations, Title 10, Part 100, Section 11 (10CFR 100.11), "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

35. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)," March 1972.

36. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.

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.~~

15.6.6 Locked Rotor from Three RCP Initial Conditions Analysis

The locked rotor accident from three RCP initial conditions analysis results are shown in Figures 15-136 through 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 9 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 1.33, which occurs at 2.2 seconds, is less than the design limit of 1.62. Consequently, DNBR margin does not exist, and 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		Natural Circulation Flowrate	Time After Reactor Trip (sec)
(MW_{th})	(% Power)	(% Full Flow)	
80	3.1	3.9	120
70	2.7	3.7	240
60	2.3	3.5	540
50	1.9	3.3	1,200
40	1.6	3.0	2,110
30	1.2	2.7	5,520
20	0.8	2.4	26,400
10	0.4	1.9	280,800

INSERT C

15.6.~~8~~ 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 result of 1.69 is equal to the DNBR limit of 1.69. The results of the locked rotor accident analyses show that the limiting locked rotor event is from a three 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 results of the

Insert C

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.

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. At 5580 seconds, the unit is cooled down to 532°F (Figure 15-156) using the ADVs on both steam lines. The ruptured steam generator is isolated after reaching 532°F (~6689 seconds), with all steam release flowpaths and EFW being isolated by 7289 seconds. After one RCP is tripped per loop, 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 (11,112 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 17,080 seconds. Boron sampling and boration to cold shutdown conditions is accomplished by 24,280 seconds, with the plant cooldown resuming at 24,580 seconds. LPIS decay heat removal conditions are reached at 41,123 seconds, where RCS pressure and temperature are held constant while this system is aligned. The plant cooldown continues at 43,823 seconds, with the RCS reaching 212°F at 48,367 seconds. The analysis is terminated at this time since steam releases to the atmosphere have stopped.

15.9.3 Environmental Consequences

~~(TO BE SUBMITTED LATER)~~ INSERT D

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.

Insert D

15.9.3 Environmental Consequences

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 analysis incorporates assumptions of 1 percent ruptured fuel and steam generator leakage prior to the postulated accident for a time sufficient to establish equilibrium specific activities in the secondary system. Two postulated cases are analyzed:

Case 1: Equilibrium RCS iodine concentrations consistent with 1%-failed fuel exist at the time of the accident. No iodine release rate spiking is assumed.

Case 2: Pre-existing iodine spike at the time the accident occurs. The reactor coolant concentrations are the maximum permitted for full power operation (50 times the normal equilibrium Technical Specification limit).

The secondary side coolant activity prior to the accident corresponds to limits set by Technical Specifications.

The initial conditions, boundary conditions and assumptions of the analysis were chosen to maximize the release of radionuclides to the environment by maximizing the stored energy in the primary system, the primary to secondary leakage, and the secondary-side pressurization. The following are used to calculate the activity release and offsite dose for SGTR accident:

1. The reactor is assumed to be at the end of a 500 EFPD cycle with extended operation at 102% full power (2619.4 MWt).

2. Prior to the accident, an equilibrium activity of fission products exists in the primary system.

3. The accident is initiated by the rupture of a steam generator tube, which results in a time-varying leakage rate of reactor coolant into the shell side of the ruptured steam generator.

4. No fuel cladding ruptures or fuel melting occur during the accident.

5. Offsite power is maintained for the duration of the accident.

6. A pre-existing 1 gpm primary to secondary leakage is assumed to be located in the unaffected steam generator. The leakage rate varies as the plant is cooled to Mode 5.

7. The steam release from the ruptured steam generator terminates at 13.5 hours, once the RCS has been cooled to Mode 5 conditions and the RCS pressure is less than or equal to the pressure in the secondary-side of the faulted OTSG. The release from the nonruptured steam generator terminates at the same time.

Insert D Continued

8. The concentrations of noble gases in the secondary side water are assumed to be negligible since any noble gases entering the secondary side are continuously vented to the atmosphere.

9. The following iodine partition coefficients are assumed:
   a. The steam generator steam/water interface iodine partition coefficient is 1.
   b. The partition coefficient for main steam/auxiliary steam providing the motive force in the condenser steam air ejectors (CSAEs) is 1.
   c. The condenser partition coefficient is 10,000 for iodine entering the condenser and then exiting via the CSAEs to the unit vent prior to reactor trip.

10. Credit is taken for the effect of radioactive decay of the RCS and secondary side radionuclide inventories during the progression of the accident.

11. For Case 1, the primary coolant concentration is at the equilibrium levels consistent for 1% failed fuel, which bounds the equilibrium Technical Specification limit.

12. For Case 2, the primary coolant concentration is at the maximum permitted for full power operation (50 times the normal equilibrium Technical Specification limit).

13. For both cases, the initial concentration of SG inventory and the feedwater are assumed to be based on 0.1 μCi/g DEI-131, the maximum concentration permitted by Technical Specification

14. Other parameters are listed in Table 15-52.

Based on the foregoing model, the thyroid and whole body doses are calculated at the exclusion area boundary and the low population zone. The results are presented in Table 15-52. The doses at these distances are below the regulatory acceptance criteria of full 10 CFR 100 limits, for each of the above cases analyzed.

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 15-3 shows the percentage of fuel pins that exceed the DNBR limit for each case. The limiting case is the 102% power case at BOC, with 40.6% of the fuel rods predicted to exceed the DNBR limit.

15.12.5 Peak RCS Pressure Analysis

The peak RCS pressure for the ARROTTA 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 82% power BOC case. Figure 15-36 shows the pressure transient, which peaks at 2885 psig at 2.3 seconds.

15.12.6 Environmental Consequences

~~(TO BE SUBMITTED LATER)~~ INSERT E

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. The limiting peak fuel pellet average enthalpy is 132.8 cal/gm. The maximum predicted fuel cladding failure percentage is 40.6%. The peak RCS pressure is 2885 psig. The environmental consequences analysis results are within the 10CFR100 limits. All of the acceptance criteria are met.

15.12.8 References

1. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components", ASME

Insert E

15.12.6 Environmental Consequences

A conservative analysis for a postulated rod ejection accident is performed to determine the resulting radiological consequences. The rod ejection accident offsite dose calculation is based on the approach provided in Regulatory Guide 1.77, with the exception that credit is taken for concurrent releases of primary coolant to the Reactor Building and secondary-side of the steam generators, when estimating doses arising from secondary system releases. During the accident, two activity release paths contribute to the total radiological consequences. 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 releases to the atmosphere. The consequences arising from these release paths are evaluated separately and then are combined to provide a total accident estimate.

The initial conditions, boundary conditions and assumptions of the analysis were chosen to maximize the release of radionuclides to the environment by maximizing the initial fission product inventories, containment fission product releases, the primary to secondary leakage, and the secondary-side coolant releases. The following are used to calculate the activity releases and offsite dose for the postulated rod ejection accident:

1. The reactor is assumed to be at the end of a 500 EFPD cycle with extended operation at 102% full power (2619.4 MWt).

2. Prior to the accident, an equilibrium RCS activity corresponding to 1 percent failed fuel exists in the primary system.

3. No fuel melting occurs.

4. The accident is initiated by the ejection of a control rod element, resulting in a breach in the reactor vessel head.

5. Fifty percent (50%) of the fuel rods in the core are analyzed to fail due to DNB, releasing all stored gap activity. Due to the magnitude of the resulting RCS fission product activities, no iodine release rate spiking is assumed.

The following are used in the analysis of the containment path releases of radioactivity to the environment:

6. The containment leak rate is 0.25 percent per day for the first 24 hours, and 50 percent of this rate thereafter.

7. The ECCS leakage during the sump recirculation is assumed to be 2 gallons per hour. There is no ECCS leakage during the injection phase.

Insert E Continued

8. For evaluation of non-ECCS releases from the RB, all (100%) of the noble gases and a quarter (25%) of the RCS coolant and fuel pin gap iodine inventories are assumed to be released to the containment atmosphere.

9. For evaluation of ECCS-related releases from the RB, all (100%) of the RCS coolant and fuel pin gap iodine inventories are assumed to be deposited in the RB sump water.

10. The iodine chemical species fractions are 0.91 elemental, 0.05 particulate, and 0.04 organic.

11. Fifty percent of the Containment leakage bypasses the Penetration Room Ventilation System (PRVS) filters.

12. The PRVS filter removal efficiencies are 90% for elemental iodines, 90% for particulate iodines, and 70% for organic iodine species.

13. No credit is taken for iodine removal from the containment atmosphere by the Reactor Building sprays.

14. Credit is taken for the effect of radioactive decay of the RCS and secondary-side radionuclide inventories during the progression of the accident.

15. To maximize ECCS leakage releases, the switchover to the Reactor Building sump recirculation mode is conservatively assumed to begin 30 minutes into the accident.

16. The assumed volume of water in the sump post-accident is 45,724 cubic feet.

17. The flashing fraction of the ECCS leakage is assumed to be 0.05.

The following are used in the analysis of the secondary system releases of radioactivity to the environment:

18. The initial concentration of SG inventory and the feedwater are assumed to be equivalent to 0.1 μCi/gm DEI-131.

19. All (100 percent) of the noble gases and iodines present in the fuel gap regions of the 50% failed fuel is mixed instantaneously within the entire reactor coolant volume, with the pre-existing RCS activity.

20. During the plant cool-down, credit is taken for depletion of the RCS fission product activities by radioactive decay and by releases to containment through the breach in the RCS pressure boundary created by the ejected rod.

21. The initial primary to secondary leak rate is 300 gal/day, distributed equally between the two OTSGs.

Insert E Continued

22. The following iodine partition coefficients are assumed:

    a) The steam generator steam/water interface iodine partition coefficient is 1.

    b) The partition coefficient for main steam/auxiliary steam providing the motive force in the condenser steam air ejectors (CSAEs) is 1.
    c) The condenser partition coefficient is 10,000 for iodine entering the condenser and then exiting via the CSAEs to the unit vent prior to reactor trip.

23. All noble gases within the RCS coolant released to the OTSG secondary sides are immediately released to the atmosphere, with no retention in the secondary coolant.

24. Offsite power is maintained for the duration of the accident.

25. Other parameters are listed in Table 15-53.

Based on the foregoing model, the primary and secondary side releases may be calculated, as well as the offsite doses. The doses corresponding to containment releases, to secondary-side releases, and the composite accident doses are given in Table 15-53. The offsite doses for the postulated rod ejection accident are below the regulatory acceptance criteria of full 10 CFR 100 limits for 0-2 hour exclusion area boundary and the outer boundary of the low population zone.

15.13.3 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 10 seconds. Due to the loss of power, the MFW pumps will begin to coast down due to loss of the condensate booster pumps. The results of the analysis are shown in Figures 15-161 through 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 repressurizes and opens the turbine bypass valves. The affected steam generator has depressurized to about 400 psig by the end of the analysis. The break flow response is similar to the with offsite power analysis. The cooldown in the affected loop is much more severe than in the unaffected loop, 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 stabilizes.

The system analysis results are input to the detailed core thermal-hydraulic analysis to determine the limiting DNBR. The minimum DNBR assuming a standard reference power distribution is 1.45, which is less than the design limit of 1.50. Consequently, DNBR margin may not exist, and 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 steam line break accident without offsite power is that DNB margin exists for all of the fuel pins.

15.13.4 Environmental Consequences

~~(TO BE SUBMITTED LATER)~~ INSERT F

15.13.5 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 10CFR100 limits. All of the acceptance criteria are met.

Insert F

15.13.4 Environmental Consequences

A conservative consequences analysis is performed for the postulated double-ended break of a 36-inch main steam line. The rapid cool-down of the Reactor Coolant System (RCS) and the associated positive reactivity addition to the core do not lead to accident-induced fuel failures or breaches of the primary system pressure boundary. Therefore, environmental consequences can only arise from atmospheric releases of pre-existing RCS activity via primary-to-secondary leakage and of contaminated secondary system coolant.

Two postulated cases are analyzed:

Case 1: Equilibrium RCS iodine specific concentrations, at the Technical Specification limit, exist at the time of the accident. The primary activities for non-iodine isotopes bound the limits set by Technical Specifications. No iodine release rate spiking is assumed.

Case 2: Pre-existing iodine spike at the time the accident occurs. The reactor coolant concentrations are the maximum permitted for full power operation (50 times the normal equilibrium Technical Specification limit).

The secondary side coolant activity prior to the accident corresponds to limits set by Technical Specifications.

The initial conditions, boundary conditions and assumptions of the analysis are chosen to maximize the release of radionuclides to the environment by maximizing the primary to secondary leakage and the secondary-side steam release. The following assumptions and parameters are used to calculate the activity release and offsite dose for the postulated large MLSB accident:

1. The reactor is assumed to be at the end of a 500 EFPD cycle with extended operation at 102% full power (2619.4 MWt).

2. No cladding ruptures or fuel melting occur during the accident.

3. Offsite power is maintained for the duration of the accident.

4. A pre-existing, 1 gpm primary-to-secondary leakage is assumed to be located in the affected steam generator. This leakage is assumed to vary with time as the plant is cooled to Mode 5. The associated activity releases to the environment bound those corresponding to primary-to-secondary leakage at the Technical Specification limits.

5. During the accident, credit is taken for depletion of the RCS fission product activities by radioactive decay and by primary-to-secondary leakage.

6. All noble gases contained in the RCS coolant leaking to the OTSG secondary side are immediately released to the atmosphere, with no retention in the secondary coolant.

Insert F Continued

7. The following iodine partition coefficients are assumed:

   a) The steam generator steam/water interface iodine partition coefficient is 1.
   b) The partition coefficient for main steam/auxiliary steam providing the motive force in the condenser steam air ejectors (CSAEs) is 1.
   c) The condenser partition coefficient is 10,000 for iodine entering the condenser and then exiting via the CSAEs to the unit vent prior to reactor trip.

8. No credit is taken for either heat removal or iodine retention in the condenser once the turbine stop valves close. Instead, the atmospheric dump valves associated with the unruptured main steam line is used to cool the plant to Mode 5.

9. Beginning at 70 minutes, the plant is cooled to Mode 5 at a rate within the Technical Specification limits that leads to conservative offsite doses.

10. SG steaming to the atmosphere ends at 2.4 hours with primary to secondary releases ending at 14.3 hours.

11. The primary coolant releases from the ruptured steam generator terminates once the RCS has been cooled to Mode 5 conditions and the RCS pressure is less than or equal to the pressure in the secondary-side of the OTSGs.

12. The RCS is assumed to be cooled to Mode 5 conditions at the Technical Specification maximum permissible rates. This maximizes the offsite dose consequences, since the secondary inventory release dominates the dose result.

13. No credit is taken for plate-out of iodine in the steam generator or steam lines.

14. Other parameters are listed in Table 15-54.

Prior to reactor trip and closure of the turbine stop valves at 0.8 seconds, primary coolant and contaminated steam are released to the condenser and through the break by both steam generators. Following closure of the turbine stop valves, the OTSG on the unaffected steam line will be isolated from the faulted OTSG. Subsequent atmospheric releases of fission products from the unfaulted OTSG occur primarily from steam releases via the associated main steam safety valves (MSSVs) or atmospheric dump valve (ADV). Post-trip primary-to-secondary leakage in the faulted OTSG is assumed to be released directly to the environment until the plant is brought to Mode 5, and the RCS is depressurized. By 70 minutes, steam flow from the unfaulted OTSG to the station Auxiliary Steam System and the turbine driven EFW pump is assumed to be isolated.

Based on the foregoing model, the primary and secondary side releases may be calculated, as well as the offsite doses. The doses are below the regulatory acceptance criteria of full 10 CFR 100 limits for each of the above cases. The results are presented in Table 15-54.

15.17.2 Analysis

The limiting small steam line break accident for DNB considerations is a break size of 1.2 ft^2 initiated from three RCP operation, with a moderator temperature coefficient of -12 pcm/°F. The transient response is given in Figures 15-168 through 15-173 and the sequence of events is given in Table 15-49. The duration of the analysis is 250 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 30% (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. As the power level increases, the temperature increases and the moderator and Doppler feedback mitigates the power excursion. The transient reaches a sustained power level of approximately 113%. 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. The system analysis results are input to a detailed core thermal-hydraulic analysis assuming a standard reference power distribution. The minimum DNBR of 1.301 is less than the design limit of 1.53. Consequently, DNBR margin may not exist, and 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 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 counted in the source term for the offsite dose calculation. 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. The centerline fuel melt limit has been evaluated and it is not violated.

15.17.3 Environmental Consequences

~~(TO BE SUBMITTED LATER)~~ INSERT G

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.

Insert G

15.17.3 Environmental Consequences

A conservative consequences analysis is performed for the postulated small steam line break outside containment. The transient cool-down of the Reactor Coolant System (RCS) and the associated positive reactivity addition to the core do not lead to accident-induced fuel failures or breaches of the primary system pressure boundary. Therefore, environmental consequences can only arise from atmospheric releases of pre-existing RCS activity via primary-to-secondary leakage and of contaminated secondary system coolant.

Two postulated cases are analyzed:

Case 1: Equilibrium RCS iodine specific concentrations, at the Technical Specification limit, exist at the time of the accident. The primary activities for non-iodine isotopes bound the limits set by Technical Specifications. No iodine release rate spiking is assumed.

Case 2: Pre-existing iodine spike at the time the accident occurs. The reactor coolant concentrations are the maximum permitted for full power operation (50 times the normal equilibrium Technical Specification limit).

The secondary side coolant activity prior to the accident corresponds to limits set by Technical Specifications.

The initial conditions, boundary conditions and assumptions of the analysis are chosen to maximize the release of radionuclides to the environment by maximizing the primary to secondary leakage and the secondary-side steam release. The following assumptions and parameters are used to calculate the activity release and offsite dose for the postulated small steam line break accident:

1. The reactor is assumed to be at the end of a 500 EFPD cycle with extended operation at 102% full power (2619.4 MWt).

2. No cladding ruptures or fuel melting occur during the accident.

3. Offsite power is maintained for the duration of the accident.

4. A pre-existing, 1 gpm primary-to-secondary leakage is assumed to be located in the affected steam generator. This leakage is assumed to vary with time as the plant is cooled to Mode 5. The associated activity releases to the environment bound those corresponding to primary-to-secondary leakage at the Technical Specification limits.

5. During the accident, credit is taken for depletion of the RCS fission product activities by radioactive decay and by primary-to-secondary leakage.

6. All noble gases contained in the RCS coolant leaking to the OTSG secondary side are immediately released to the atmosphere, with no retention in the secondary coolant.

Insert G Continued

7. The following iodine partition coefficients are assumed:

    a) The steam generator steam/water interface iodine partition coefficient is 1.
    b) The partition coefficient for main steam/auxiliary steam providing the motive force in the condenser steam air ejectors (CSAEs) is 1.
    c) The condenser partition coefficient is 10,000 for iodine entering the condenser and then exiting via the CSAEs to the unit vent prior to reactor trip.

8. No credit is taken for either heat removal or iodine retention in the condenser once the turbine stop valves close. Instead, the atmospheric dump valves associated with the unruptured main steam line are used to cool the plant to Mode 5.

9. Beginning at 70 minutes, the plant is cooled to Mode 5 at a rate within the Technical Specification limits that leads to conservative offsite doses.

10. SG steaming to atmosphere ends at 2.9 hours with primary to secondary releases ending at 3.9 hours.

11. The RCS is assumed to be cooled to Mode 5 conditions at the Technical Specification maximum permissible rates. This maximizes the offsite dose consequences, since the secondary inventory release dominates the dose result.

12. No credit is taken for plate-out of iodine in the steam generator steam lines.

13. The primary coolant releases from the ruptured steam generator terminate once the RCS has been cooled to Mode 5 conditions and the RCS pressure is less than or equal to the pressure in the secondary-side of the OTSGs.

14. Other parameters are listed in Table 15-55.

Prior to a manual reactor trip at 10 minutes, the RCS depressurizes and begins cooldown leading to positive reactivity insertion and an increase in core power. Following the manual reactor trip, one motor-driven emergency feedwater pump starts and motive steam flow to the turbine-driven emergency feedwater pump is initiated. The operators stabilize the plant and hold the plant at the thermal shock operating region for 1 hour and begin a controlled cooldown at 70 minutes.

Based on the foregoing model, the primary and secondary side releases may be calculated, as well as the offsite doses. The doses are below the regulatory acceptance criteria of a small fraction (10%) of 10 CFR 100 limits, for each of the above cases analyzed. The results are presented in Table 15-55.

Table 15-16 Summary of Transient and Accident Doses Including the Effects of High Burnup Reload Cores

		Doses (rem)	
Fuel Handling Accident			
	Thyroid at EAB	5.25E+1	
	Whole body at EAB	1.85E-1	
Steam Generator Tube Rupture		**Case 1**	**Case 2**
	Thyroid at EAB	3.95E+1	2.62E+2
	Whole body at EAB	1.52E-1	2.57E+2
	Thyroid at LPZ	1.13E+1	6.61E+1
	Whole body at LPZ	2.80E-2	4.16E-1
Waste Gas Tank Failure			
	Thyroid at EAB	2.70E-1	
	Whole body at EAB	1.70E-1	
Rod Ejection			
	Thyroid at EAB	8.46E+1	
	Whole body at EAB	3.51E-1	
	Thyroid at LPZ	1.15E+1	
	Whole body at LPZ	4.00E-2	
Large Main Steam Line Break		**Case 1**	**Case 2**
	Thyroid at EAB	9.80E+0	1.14E+1
	Whole body at EAB	2.33E-3	1.46E-2
	Thyroid at LPZ	1.08E+0	1.58E+0
	Whole body at LPZ	2.66E-4	2.64E-3
Small Main Steam Line Break		**Case 1**	**Case 2**
	Thyroid at EAB	4.97E+0	6.69E+0
	Whole body at EAB	2.23E-3	1.53E-2
	Thyroid at LPZ	6.17E-1	1.52E+0
	Whole body at LPZ	3.04E-4	4.33EE-3
LOCA			
	Thyroid at EAB	5.00E+0	
	Whole body at EAB	1.00E-2	
	Thyroid at LPZ	5.50E+0	
	Whole body at LPZ	1.40E-2	
Maximum Hypothetical Accident			
	Thyroid at EAB	2.64E+2	
	Whole body at EAB	1.62E+0	
	Thyroid at LPZ	1.29E+2	
	Whole body at LPZ	3.83E-1	

Table 15-16	Summary of Transient and Accident Doses Including the Effects of High Burnup Reload Cores	
Fuel shipping cask drop accident*		
	Thyroid at EAB	1.42E+2
	Whole body at EAB	1.30E-1
Dry storage transfer cask drop*		
	Thyroid at EAB	7.20E+1
	Whole body at EAB	1.50E-1
* Not reanalyzed for high burnup fuel, although the impact is negligible.		

Table 15-50. Iodine and Noble Gas Inventory in Reactor Core and Fuel Rod Gaps - 500 EFPD Equilibrium Cycle[1]

Nuclide	Core Inventory (Curies)	Fraction of Inventory in Gap[2,3]	Total Core Gap Inventory (Curies)
I-131	7.16E+7	0.10	7.16E+6
I-132	1.03E+8	0.10	1.03E+7
I-133	1.49E+8	0.10	1.49E+7
I-134	1.73E+8	0.10	1.73E+7
I-135	1.35E+8	0.10	1.35E+7
Xe-131m	4.51E+6	0.10	7.89E+4
Xe-133m	4.51E+6	0.10	4.51E+5
Xe-133	1.39E+8	0.10	1.39E+7
Xe-135m	2.82E+7	0.10	2.82E+6
Xe-135	4.17E+7	0.10	4.17E+6
Xe-138	1.19E+8	0.10	1.19E+7
Kr-83m	8.95E+6	0.10	8.95E+5
Kr-85m	1.90E+7	0.10	1.90E+6
Kr-85	8.30E+5	0.30	2.49E+5
Kr-87	3.66E+7	0.10	3.66E+6
Kr-88	5.15E+7	0.10	5.15E+6

Notes:

1. Based on an equilibrium cycle at end of life. The three-region core operates at a power level of 2619.36 MWth (i.e., 102% of full power).
2. Regulatory Guide 1.25.
3. Regulatory Guide 1.77, Appendix B.

Table 15-51 Reactor Coolant System Fission Product Source Activities - 500 EPFD Equilibrium Cycle [1]

Nuclide	Maximum Specific Activity [2] (μCi/gm)	Maximum Total Coolant Activity [3] (Curies)
I-131	5.80E+0	1.39E+3
I-132	8.36E+0	1.68E+3
I-133	7.03E+0	1.42E+3
I-134	7.76E-1	1.55E+2
I-135	3.32E+0	6.63E+2
Xe-131m	6.04E+0	1.21E+3
Xe-133m	6.42E+0	1.28E+3
Xe-133	4.67E+2	9.32E+4
Xe-135m	7.06E-1	1.41E+2
Xe-135	1.34E+1	2.67E+3
Xe-138	7.40E-1	1.48E+2
Kr-83m	5.34E-1	1.06E+2
Kr-85m	2.23E+0	4.44E+2
Kr-85	1.72E+2	3.43E+4
Kr-87	1.21E+0	2.42E+2
Kr-88	3.81E+0	7.60E+2

Notes:

1. Reactor coolant activities at equilibrium assuming 1 percent failed fuel randomly distributed throughout the core.
2. Based on steady-state operation with no RCS leakage and no continuous pressurizer spray flow. Used for calculating doses arising from reactor coolant leaks to the secondary systems.
3. Based on steady-state operation with no RCS leakage and 1 gpm continuous pressurizer spray flow. Used for calculating rod ejection activity releases to containment.

Table 15-52: Parameters for Steam Generator Tube Rupture Accident Dose Analysis

			Case 1	Case 2
1.	**Data and assumptions used to estimate radioactive sources from postulated accident.**			
	a.	Power level (MWth)	2619.4	Same
	b.	Number of failed fuel rods	368	234.3
	c.	Initial primary to secondary leak rate (gpm)	1.0	Same
2.	**Data and assumptions used to estimate activity released**			
	a.	Iodine partition coefficient for steam releases from ruptured steam generator	1.0	Same
	b.	Iodine partition coefficient for steam releases from unruptured steam generator	1.0	Same
	c.	Iodine partition coefficient in condenser	10,000	Same
	d.	Reactor coolant released to ruptured steam generator (lbm)	1.076E+6	Same
	e.	Reactor coolant released to unruptured steam generator (lbm)	3.018E+3	Same
3.	**Dispersion data**			
	a.	Distance to exclusion area boundary (m)	1.609E+3	Same
	b.	Distance to outer boundary of low population zone (m)	9.654E+3	Same
	c.	Chi/Q at exclusion area boundary (sec/m3)		
		0-2 hrs (unit vent releases)	3.35E-5	Same
		0-2 hrs (all other releases)	2.20E-4	Same
	d.	Chi/Q at low population zone boundary (sec/m3)		
		0-8 hrs	2.35E-5	Same
		8-24 hrs	4.70E-6	Same
		1-4 days	1.50E-6	Same
		4-30 days	3.30E-7	Same
4.	**Dose data**			
	a.	Method of dose calculations	Reg. Guide 1.4	Same
	b.	Dose conversion assumptions	ICRP 30 NUREG-1918	Same
	c.	Doses (rem)	see Table 15-16	

Table 15-53: Parameters for Postulated Rod Ejection Offsite Dose Analysis

1.	**Data and assumptions used to estimate radioactive sources from postulated accident.**		
	a.	Power level (MWth)	2619.4
	b.	Percent of failed fuel prior to accident	1.0
	c.	Initial primary to secondary leak rate (gpd)	150 gpd in each OTSG
	d.	Failed fuel	50 percent of fuel rods in core
	e.	Activity released to reactor coolant from failed fuel and available for release.	
		Noble gases	80 percent of core gap inventory
		Iodines	80 percent of core gap inventory
	f.	Iodine species fractions (organic, elemental, and particulate)	Regulatory Guide 1.4
2.	**Data and assumptions used to estimate activity released**		
	a.	Containment free volume (ft^3)	1.828E+6
	b	Containment leak rate	0.25 percent of containment volume per day, $0 \le t \le 24$ hrs. 0.125 percent of containment volume per day, t>24 hrs.
	c.	Bypass leakage fraction	0.50
	d.	Iodine partition coefficient for steam releases	1.0
	e.	Iodine partition coefficient in condenser	10,000
	f.	Primary coolant releases to Reactor Building (lbs)	
		(0-2 hrs)	1,190,439
		(0-8 hrs)	4,576,102
		(0-24 hrs)	--
		(0-4 days)	--
		(0-30 days)	--
	h.	Primary coolant leakage to secondary-side of OTSGs (lbs)	
		(0-2 hrs)	71.3
		(0-8 hrs)	314.0
		(0-24 hrs)	1000.8
		(0-4 days)	4116.4
		(0-30 days)	8929.7
	i.	Steam release from steam generators (lbs)	
		(0-2 hrs)	298,058
		(0-8 hrs)	1,166,649
		(0-24 hrs)	1,719,594
		(0-4 days)	2,735,614
		(0-30 days)	4,305,292
3.	**Dispersion data**		
	a.	Distance to exclusion area boundary (m)	1.609E+3
	b.	Distance to outer boundary of low population zone (m)	9.654E+3
	c.	Chi/Q at exclusion area boundary (sec/m3)	

Table 15-53: Parameters for Postulated Rod Ejection Offsite Dose Analysis

	0-2 hrs (unit vent releases)	3.35E-5
	0-2 hrs (all other releases)	2.20E-4
d.	Chi/Q at low population zone boundary (sec/m3)	
	0-8 hrs	2.35E-5
	8-24 hrs	4.70E-6
	1-4 days	1.50E-6
	4-30 days	3.30E-7
4.	**Dose data**	
a.	Method of dose calculations	Regulatory Guide 1.77, with the following exception: Credit is taken for concurrent RCS releases to RB and primary-to-secondary leakage.
b.	Dose conversion assumptions	ICRP 30 NUREG-1918
c.	Doses (rem)	see Table 15-16

Table 15-54: Parameters for Large Main Steam Line Break Accident Dose Analysis

			Case 1	Case 2
1.		**Data and assumptions used to estimate radioactive sources from postulated accident.**		
	a.	Power level (MWth)	2619.4	2619.4
	b.	RCS iodine specific activity [μCi/gm]	1.0	50 times Case 1
	c.	Non-iodine RCS activity	Tech Spec limit	50 times Case 1
	d.	Faulted OTSG tube initial leak rate (gpm)	1.0	Same
	e.	Unfaulted OTSG tube initial leak rate (gpd)	150	150
2.		**Data and assumptions used to estimate activity released**		
	a.	Iodine partition coefficient for steam releases	1.0	1.0
	b.	Iodine partition coefficient in condenser	10,000	10,000
	c.	Integrated steam release from steam generators (lbm)		
		1200 sec.	45,920	Same
		4500 sec.	267,359	Same
		7200 sec.	448,779	Same
3.		**Dispersion data**		
	a.	Distance to exclusion area boundary (m)	1.609E+3	Same
	b.	Distance to outer boundary of low population zone (m)	9.654E+3	Same
	c.	Chi/Q at exclusion area boundary (sec/m3)		
		0-2 hrs (unit vent releases)	3.35E-5	Same
		0-2 hrs (all other releases)	2.20E-4	Same
	d.	Chi/Q at low population zone boundary (sec/m3)		
		0-8 hrs	2.35E-5	Same
		8-24 hrs	4.70E-6	Same
		1-4 days	1.50E-6	Same
		4-30 days	3.30E-7	Same
4.		**Dose data**		
	a.	Method of dose calculations	Reg. Guide 1.4	Same
	b.	Dose conversion assumptions	ICRP 30 NUREG-1918	Same
	c.	Doses (rem)	see Table 15-16	

Table 15-55: Parameters for Small Main Steam Line Break Accident Dose Analysis

		Case 1	Case 2
1.	**Data and assumptions used to estimate radioactive sources from postulated accident.**		
	a. Power level (MWth)	2619.4	Same
	b. RCS iodine specific activity [μCi/gm]	1.0	50 times Case 1
	c. Non-iodine RCS activity	Tech Spec limit	50 times Case 1
	d. Faulted OTSG tube initial leak rate (gpm)	1.0	Same
	e. Unfaulted OTSG tube initial leak rate (gpm)	150 gpd	Same
2.	**Data and assumptions used to estimate activity released**		
	a. Iodine partition coefficient for steam releases	1.0	Same
	b. Iodine partition coefficient in condenser	10,000	Same
	c. Integrated steam release from steam generators (lbm)		
	1200 sec.	49,216	Same
	4500 sec.	263,318	Same
	7200 sec.	416,196	Same
3.	**Dispersion data**		
	a. Distance to exclusion area boundary (m)	1.609E+3	Same
	b. Distance to outer boundary of low population zone (m)	9.654E+3	Same
	c. Chi/Q at exclusion area boundary (sec/m3)		
	0-2 hrs (unit vent releases)	3.35E-5	Same
	0-2 hrs (all other releases)	2.20E-4	Same
	d. Chi/Q at low population zone boundary (sec/m3)		
	0-8 hrs	2.35E-5	Same
	8-24 hrs	4.70E-6	Same
	1-4 days	1.50E-6	Same
	4-30 days	3.30E-7	Same
4.	**Dose data**		
	a. Method of dose calculations	Regulatory Guide 1.4	Same
	b. Dose conversion assumptions	ICRP 30 NUREG-1918	Same
	c. Doses (rem)	see Table 15-16	

ATTACHMENT 2
DESCRIPTION OF THE PROPOSED CHANGES AND
TECHNICAL JUSTIFICATIONS FOR THE PROPOSED CHANGES

Background

On April 5, 1999, Duke Energy Corporation (Duke) submitted a license amendment request (LAR) for Facility Operating License Nos. DPR-38, DPR-47, and DPR-55 for Oconee Nuclear Station Units 1, 2, and 3, respectively, pursuant to 10 CFR 50.90. The LAR proposed changes to the Technical Specifications, Bases, Updated Final Safety Analysis Report (UFSAR), and Core Operating Limits Report (COLR) to incorporate Topical Report DPC-NE-3005-P, "Thermal-Hydraulic Transient Analysis Methodology." It supports the future operation of Oconee Unit 2 during Cycle 18 (O2C18).

Oconee Unit 2 Cycle 18 will implement the revised UFSAR Chapter 15 non-LOCA analysis methodology of topical report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology." Revision 0 of this topical report was conditionally approved by the NRC Safety Evaluation Report (SER) dated October 1, 1998. Revision 1 was submitted on February 1, 1999, to respond to the conditions in the SER. Additional information regarding Revision 1 of DPC-NE-3005-P was submitted on April 19 and May 5, 1999. The UFSAR Chapter 15 revisions are based on the Revision 1 methodology. Since the methodology is being separately reviewed and the analysis results presented follow the methodology, the UFSAR revisions will be the implementation of an approved methodology once Revision 1 is approved by the NRC. Many of the results in the UFSAR revisions were already presented in Revisions 0 and 1 of DPC-NE-3005-P.

The environmental consequences (offsite dose analysis) associated with the revised Chapter 15 analyses were not submitted on April 5, 1999. Additionally, the environmental consequences analysis methodology is not included in Topical Report DPC-NE-3005. Duke committed to submit the dose methodology analysis and associated UFSAR marked pages to the NRC by May 31, 1999. This submittal provides the revised dose analyses and the associated UFSAR changes.

Description and Justification of the UFSAR Changes

Mark-ups of the UFSAR are provided in Attachment 1. These mark-ups reflect the new dose analyses. In general the revised UFSAR pages provided in the April 5, 1999, submittal have been marked up to document the inclusion of the environmental consequences. However, the current UFSAR has been marked-up to show the changes to UFSAR Tables 2-29 and 2-30.

The dose analyses regarding a Steam Generator Tube Rupture (SGTR), Rod Ejection Accident (REA), and Main Steam Line Break (MSLB) were recalculated. Changes in the dose results arose from changes in the following initial boundary conditions:

1) The Reactor Coolant System (RCS) and core source terms are based on cycle designs based on 500 Effective Full Power Days (EFPD), not 400 EFPD cycles. This increases the dose by ~10-20%.

2) Secondary-side fission product activities are higher than the values used in the UFSAR analysis. This results in higher doses. The revised calculations consistently apply the technical specification values.

3) International Commission on Radiological Protection (ICRP) 30 dose conversion factors for iodine isotopes are used in place of ICRP 2 values to reflect advances in scientific knowledge. This decreases the dose by ~ 20%.

4) The Units 2 and 3 ground release X/Q (2.2E-4sec/m3) is used in lieu of the current Unit 1 value. This increases doses by ~ 90%. The Unit 1 value was calculated prior to the lake being filled. The Units 2 and 3 ground release X/Q more accurately reflects the current conditions.

5) Integrated RCS coolant and contaminated feedwater releases to the environment are predicted to be significantly greater than those assumed in the current UFSAR analysis.

6) An iodine partition coefficient of 1 is assumed per past NRC input to the B&W Owner's Group.

7) No credit is taken for post-trip availability of the condenser. This results in increased dose.

Specifically, the following changes have been made to the above mentioned accident analyses:

<u>SGTR</u>

1) Ground release X/Qs were used instead of elevated release X/Qs to better reflect fission product release paths.

2) The accident progression provided in topical report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology," differs from the current UFSAR analysis.

<u>MSLB</u>

1) A new analysis to address small MSLBs was added in topical report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology."

2) The accident progression provided in topical report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology," differs from the current UFSAR analysis.

<u>REA</u>

1) The accident-induced fuel pin failures is increased from 28% to 50% per new core design and methodology analysis.

2) RCS fission product releases via SG tube leakage is modeled.

3) The iodine fuel pin gap fraction is increased from 2% to 10% in accordance with Regulatory Guide 1.77.

Conclusions

The revised dose analyses regarding a SGTR, REA, and MSLB continue to demonstrate that the resultant doses remain within the post-accident acceptance criteria (i.e., 10 CFR 100 limits).

ATTACHMENT 3

NO SIGNIFICANT HAZARDS CONSIDERATION

Pursuant to 10 CFR 50.91, Duke has made the determination that this License Amendment Request involves No Significant Hazards by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated?

   No. The proposed changes to the Technical Specifications, Bases, Updated Final Safety Analysis Report (UFSAR), and Core Operating Limits Report (COLR) incorporate the accident analyses established in Topical Report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology." On July 30, 1997, Duke submitted Topical Report DPC-NE-3005-P to the NRC for approval. The NRC found DPC-NE-3005-P acceptable, with noted exceptions, in a Safety Evaluation issued on October 1, 1998. To resolve the noted NRC exceptions, Duke submitted Revision 1 of DPC-NE-3005-P to the NRC for review on February 1, 1999. Additional information regarding Revision 1 of DPC-NE-3005-P was submitted on April 19 and May 5, 1999. This LAR is dependent upon the NRC approval of Revision 1 of DPC-NE-3005-P.

   The analyzed events are initiated by the failure of specific plant structures, systems or components. These proposed changes do not impact the condition or performance of those structures, systems or components.

   The revised accident analyses in DPC-NE-3005-P demonstrate that the applicable acceptance criteria are met. In addition, the calculations show that the applicable radiological and environmental acceptance criteria continue to be met.

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

2. Create the possibility of a new or different kind of accident from any accident previously evaluated?

No. The proposed changes do not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. Where setpoints and operating limits have been revised, the revised accident analyses demonstrate that the applicable acceptance criteria are met. As a result, no new failure modes are being introduced.

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

3. Involve a significant reduction in a margin of safety?

No. The margin of safety is established through the design of the plant structures, systems and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The proposed changes do not involve a physical alteration of the plant. No new or different equipment is being installed, and no installed equipment is being operated in a new or different manner. Where setpoints and operating limits have been revised, the revised accident analyses in DPC-NE-3005-P demonstrate that the applicable acceptance criteria are met.

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

Conclusion

Based upon the preceding evaluation, performed pursuant to 10 CFR 50.92, Duke has concluded that the proposed changes to the Oconee Nuclear Station Technical Specifications, Bases, UFSAR, and O2C18 COLR will not involve a significant hazards consideration.

ATTACHMENT 4

ENVIRONMENTAL ASSESSMENT

Pursuant to 10 CFR 51.22(b), an evaluation of this LAR is being performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) or 10 CFR 51.22(c)(10) of the regulations.

This LAR for the Oconee Technical Specifications proposes changes to allow implementation of reactor fuel cycle 18 on Unit 2 (O2C18). The reload design for O2C18 was accomplished using DPC-NE-3005-P. The NRC approved the topical report, with some exceptions, in a Safety Evaluation Report issued on October 1, 1998. To resolve these exceptions, Duke submitted Revision 1 to DPC-NE-3005-P to the NRC on February 1, 1999. Additional information regarding Revision 1 of DPC-NE-3005-P was submitted on April 19 and May 5, 1999.

If it can be determined there are:

1) No significant hazards considerations;

2) No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite; and

3) No significant increase in individual or cumulative occupational radiation exposures involved;

then this LAR will qualify for categorical exclusion from the requirement to perform an environmental assessment/impact statement.

For Item 1, as listed above, Attachment 3 of this reload submittal package documents the determination of no significant hazards.

The associated dose analysis show that the O2C18 reload will continue to meet the applicable radiological and environmental acceptance criteria; thus, this LAR will meet the criteria for categorical exclusion from performing an environmental assessment/impact statement.