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Normal Operating Pressure/Normal Operating Temperature Program – Licensing Report"
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Westinghouse

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D. C. Cook Unit 1 Return to Reactor Coolant System Normal Operating Pressure/Normal Operating Temperature Program – Licensing Report

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LIST OF ACRONYMS AND ABBREVIATIONS

ADAMS	Agencywide Documents Access and Management System
AEP	American Electric Power
AFW	auxiliary feedwater
ANS	American Nuclear Society
AOR	analysis of record
ASTRUM	Automated Statistical Treatment of Uncertainty Method
ATWS	anticipated transient without scram
BAPC	boric acid precipitation control
BE	best-estimate
BOC	beginning of cycle
BOL	beginning of life
BWI	Babcock & Wilcox International
CAOC	constant axial offset control
CEQ	containment air recirculation/hydrogen skimmer system
CFR	Code of Federal Regulations
COLR	Core Operating Limit Report
CPT	critical power trajectories
CTS	containment spray
DER	double-ended rupture
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
ECCS	emergency core cooling system
EDG	emergency diesel generator
ELI	excessive load increase
EM	evaluation model
EOP	Emergency Operating Procedure
ESFAS	engineered safety feature actuation system
FA	fuel assembly
FWM	feedwater malfunction
GDC	General Design Criteria
HFP	hot full power
HTM	heat transfer multiplier
HZP	hot zero power

LIST OF ACRONYMS AND ABBREVIATIONS (cont.)

IM	Indiana Michigan Power Company
IFBA	integral fuel burnable absorber
IFM	intermediate flow mixing vanes
LBLOCA	large-break loss-of-coolant accident
LOCA	loss-of-coolant accident
LOF	loss of flow
LOL	loss of load
LONF/LOAC	loss of normal feedwater/loss of AC power
LOOP	loss of offsite power
LSP	low steam pressure
LTC	long-term cooling
M&E	mass and energy
MFIV	main feedwater isolation valve
MMF	minimum measured flow
MSIV	main steam isolation valve
MSLB	main steamline (steam line) break
MTO	margin to overfill
MUR	measurement uncertainty recapture
ND	nuclear design
non-LOCA	non-loss-of-coolant accident
NOP	normal operating pressure
NOT	normal operating temperature
NSAL	Nuclear Safety Advisory Letter
NSSS	nuclear steam supply system
OFA	Optimized Fuel Assembly
OPΔT	overpower delta-T
OTΔT	overtemperature delta-T
PAD	performance analysis for design
PCT	peak cladding temperature
PIV	pressure isolation valve
PORV	power-operated relief valve
PWR	pressurized water reactor
QA	quality assurance
QMS	Quality Management Systems
RCCA	rod cluster control assembly
RCP	reactor coolant pump
RCS	reactor coolant system
RIP	rod internal pressure
RSAC	Reload Safety Analysis Checklist
RSE	Reload Safety Evaluation

LIST OF ACRONYMS AND ABBREVIATIONS (cont.)

RSG	replacement steam generator
RTDP	Revised Thermal Design Procedure
RTS	reactor trip system
RWAP	rod withdrawal at power
RWFS	rod withdrawal from subcritical
RWST	refueling water storage tank
SAFDL	specified acceptable fuel design limit
SBLOCA	small-break LOCA
SER	Safety Evaluation Report
SG	steam generator
SGTP	steam generator tube plugging
SGTR	steam generator tube rupture
SI	safety injection
SLB	steam line (steamline) break
SUIL	startup of an inactive loop
T_{avg}	reactor vessel average temperature
T_{cold}	cold leg temperature
T_{feed}	feedwater temperature
T_{hot}	hot leg temperature
TCD	thermal conductivity degradation
TDF	thermal design flow
T/H	thermal-hydraulic
TPI	thimble plugs installed
TPR	thimble plugs removed
TS	Technical Specification
UET	unfavorable exposure time
UFSAR	Updated Final Safety Analysis Report
USNRC	United States Nuclear Regulatory Commission
WC/T	WCOBRA/TRAC (Westinghouse COBRA/TRAC computer code)

1 INTRODUCTION

1.1 BACKGROUND

Cook Unit 1 was originally licensed at a reactor coolant system (RCS) operating pressure of 2250 psia, a nominal core inlet temperature of 536°F, and a nominal core outlet temperature of 599°F (see Section 1.2 of Reference 1). Table 4.2-1 of Reference 1 identified an average in-core fuel design temperature of 570.3°F. The reactor vessel average temperature (T_{avg}) for Cook Unit 1 has been revised in support of a number of License Amendments, including the 30 percent Steam Generator Tube Plugging (SGTP) Program (Reference 7), wherein a range of values from 553°F to 576.3°F was used in accident analyses and an upper end T_{avg} value of 576.3°F was approved in the Cook Unit 1 Technical Specifications (TS).

As a result of steam generator (SG) tube degradation in the late 1980s, Cook Unit 1 implemented a SG preservation program which included reducing pressurizer pressure to 2100 psia and T_{avg} to approximately 556°F. Although the Cook Unit 1 SGs were replaced in 1999, the reduced RCS operating conditions were maintained. During the refueling outage prior to Cook Unit 1 Cycle 24 operation, eddy current examinations of the SGs showed an unexpected increase in the tube wear in the vicinity of fan bar stabilizers (U-tube bend region) in each SG. Though SGTP levels are currently low at Cook Unit 1 (< 1 percent), this unexpected increase is inconsistent with industry experience, making future degradation projections difficult. The subsequent American Electric Power (AEP) root cause analysis indicated that the current Cook Unit 1 operating conditions are a likely contributor to the suspected pressure pulse phenomenon or "harmonic" vibration situation of the SG fan bars that underlies the degradation and that it could be mitigated by increasing the nominal full-power secondary-side SG pressure to approximately 800 psig. This would be accomplished by restoring the Cook Unit 1 RCS to "normal" pressure and temperature operating values.

AEP plans on implementing a return to RCS Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) conditions for Cook Unit 1 by:

- Increasing the current operating nominal full-power pressurizer pressure from 2100 psia to 2250 psia.
- Increasing the current operating nominal full-power T_{avg} from 556°F to 571°F.

Implementation of the program is planned to occur prior to Cook Unit 1 Cycle 26 startup (October 2014).

1.2 LICENSING REPORT PURPOSE AND CONTENT

The purpose of this Licensing Report is to document the analyses and evaluations performed by Westinghouse to demonstrate that the Cook Unit 1 nuclear steam supply system (NSSS) will continue to comply with its design and licensing basis with a return to RCS NOP/NOT conditions. Section 2 (NSSS Parameters) of this report discusses the NSSS design parameters that were updated as a result of the Cook Unit 1 Return to RCS NOP/NOT Program and that serve as the basis for the NSSS analyses and evaluations. Section 3 (Design Transients) addresses the design transient evaluations performed to accommodate the revised nominal operating conditions. Section 4 (NSSS Systems) presents the NSSS controls evaluations that were completed for the revised nominal operating conditions. Section 5

(NSSS Accident Analyses) provides the results of the accident analyses and evaluations (including SG tube rupture (SGTR), mass and energy (M&E) release and containment response calculations, loss-of-coolant accident (LOCA) and non-LOCA). Section 6 (Nuclear Fuel) provides the results of the fuel analyses and evaluations (including nuclear design (ND), fuel rod design, fuel thermal-hydraulic (T/H) and fuel mechanical design).

1.3 SUMMARY OF TECHNICAL SPECIFICATION CHANGES

The TS and TS Bases changes associated with the analyses and evaluations performed by Westinghouse for the Cook Unit 1 Return to RCS NOP/NOT Program are as follows:

- RCS Pressure Isolation Valve (PIV) Leakage (TS 3.4.14) – Surveillance Requirements (SR 3.4.14.1) – updated to include the new nominal RCS pressure range (2215 psig ≤ RCS Pressure ≤ 2255 psig).

  Bases (pg B 3.4.14-2) – Limited Condition for Operation – updated to include the new RCS pressure range.

- Seal Injection Flow (TS 3.5.5) – Surveillance Requirements (SR 3.5.5.1) – updated to include the new nominal RCS pressure range (2215 psig ≤ RCS Pressure ≤ 2255 psig).

  Bases (pg B 3.5.5-4) – Surveillance Requirements (SR 3.5.5.1) – deleted the low pressure operation value.

- Containment Spray System Bases (pg B 3.6.6-3) – Applicable Safety Analyses – updated the Containment Spray System total response time discussion.

- Containment Air Recirculation/Hydrogen Skimmer (CEQ) System (TS 3.6.10) – Surveillance Requirements (SR 3.6.10.1) – updated the surveillance delay time range for the containment air recirculation fans, (270 seconds ≤ CEQ fan delay ≤ 300 seconds).

  Bases (pg B 3.6.10-1) – Background – updated CEQ fans automatic start time by the Containment Pressure – High signal.

  Bases (pg B 3.6.10-2) – Applicable Safety Analyses – updated to include CEQ maximum start time.

  Bases (pgs B 3.6.10-3 & 4) – Surveillance Requirements (SR 3.6.10.1) – updated the CEQ System fan start and delay timing.

1.4 METHODOLOGY AND ACCEPTANCE CRITERIA

The NSSS analyses and evaluations for the Cook Unit 1 Return to RCS NOP/NOT Program were performed in accordance with Westinghouse quality assurance requirements defined in the Westinghouse Quality Management System (QMS) procedures, which comply with the Code of Federal Regulations (CFR) 10 CFR 50 Appendix B criteria. These analyses and evaluations are in conformance with Westinghouse and industry codes, standards and regulatory requirements applicable to Cook Unit 1. Assumptions and acceptance criteria are provided in the appropriate sections of this report.

The following analysis methodology is highlighted as being used for the Cook Unit 1 Return to RCS NOP/NOT Program.

- Best Estimate Large-Break LOCA Fuel Thermal Conductivty Degradation (TCD) Evaluation

  The United States Nuclear Regulatory Commission (USNRC) approved 2004 Automated Statisical Treatment of Uncertainty Method (ASTRUM) Evaluation Model (EM) (Reference 2) is based on the performance analysis for design (PAD) 4.0 fuel performance code (Reference 3). PAD 4.0 was licensed without explicitly considering fuel pellet TCD with burnup. Explicit modeling of fuel pellet TCD in the fuel performance code leads to changes in the fuel rod design parameters beyond beginning-of-life. Cook Units 1 and 2 have addressed fuel pellet TCD with plant-specific evaluations, using the ASTRUM uncertainty analysis methodology described in Reference 2 and the evaluation methodology described in Reference 4 to determine the estimated effect of fuel pellet TCD and peaking factor burndown. Specifically, as documented in Reference 5, PAD 4.0 + TCD fuel performance data that accounts for fuel pellet TCD is used as input for the Cook Unit 1 Best Estimate Large-Break LOCA (BE LBLOCA) fuel TCD evaluation.

  As reiterated by the USNRC in Reference 6, all future USNRC license amendment submittals associated with fuel or safety analysis are expected to address fuel pellet TCD. However, Westinghouse does not have an USNRC approved fuel performance model that addresses fuel pellet TCD. Therefore, the BE LBLOCA fuel TCD evaluation for the Cook Unit 1 Return to RCS NOP/NOT Program was performed using PAD 4.0 + TCD fuel performance data. Specifically, the PAD 4.0 + TCD fuel performance data was generated with a Cook Unit 1 plant-specific model (using an unlicensed model) that includes explicit modeling of fuel pellet TCD as described in Reference 4.

  To demonstrate compliance with the 10 CFR 50.46(b)(1) acceptance criterion concerning peak cladding temperature (PCT) when explicitly considering fuel pellet TCD and peaking factor burndown in the BE LBLOCA fuel TCD evaluation for the Cook Unit 1 Return to RCS NOP/NOT Program, design input values were used to gain offsetting margin. For the evaluation approach, the Integrated PCT was first calculated to demonstrate compliance with the 10 CFR 50.46(b)(1) criterion when the design input changes and TCD and peaking factor burndown were considered. Then, the Margin Recovery PCT was calculated, including only the design input changes.

1.5 REFERENCES

1. USNRC Safety Evaluation Report, "Safety Evaluation by the Directorate of Licensing U.S. Atomic Energy Commission in the Matter of Indiana & Michigan Electric Company and Indiana & Michigan Power Company, Donald C. Cook Nuclear Plant Units 1 and 2, Docket Nos. 50-315 and 50-316," September 11, 1973.

2. Westinghouse Report WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment Of Uncertainty Method (ASTRUM)," January 2005.

3. Westinghouse Report WCAP-15063-P-A, Revision 1 with Errata, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," July 2000.

4. Westinghouse Project Letter LTR-NRC-12-27, "Westinghouse Input Supporting Licensee Response to NRC 10 CFR 50.54(f) Letter Regarding Nuclear Fuel Thermal Conductivity Degradation (Proprietary/Non-Proprietary)," March 7, 2012.

5. AEP Letter from Joel P. Gebbie (AEP) to USNRC, AEP-NRC-2012-13, "Donald C. Cook Nuclear Plants Units 1 and 2 Response to Information Request Pursuant to 10 CFR 50.54(f) Related to the Estimated Effect on Peak Cladding Temperature Resulting from Thermal Conductivity Degradation in the Westinghouse-Furnished Realistic Emergency Core Cooling System Evaluation (TAC NO. M99899)," March 19, 2012. (Available in USNRC ADAMS under Accession Number ML12088A104.)

6. USNRC Information Notice 2009-23 Supplemental 1, "Nuclear Fuel Thermal Conductivity Degradation," October 26, 2012.

7. Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 – Issuance of Amendments RE: Increased Steam Genertor Tube Plugging Limit (TAC NOS. M92587 and M92588), March 13, 1997.

2 NSSS PARAMETERS

2.1 NSSS DESIGN PARAMETERS

2.1.1 Introduction and Background

The nuclear steam supply system (NSSS) design parameters are the fundamental parameters used as input in the NSSS analyses. They provide the primary-side and secondary-side system conditions (thermal power, temperatures, pressures, and flows) used as the basis for all of the NSSS analyses and evaluations. The Cook Unit 1 NSSS design parameters have been revised for the Return to Reactor Coolant System (RCS) Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) Program as shown in Tables 2.1-1 and 2.1-2.

Table 2.1-1 provides information for the four design cases associated with the Return to RCS NOP/NOT Program for Cook Unit 1. Table 2.1-2 provides information for the same four cases, but considers a reduced steam generator tube plugging (SGTP) level and increased thermal design flow (TDF). These parameters have been incorporated, as required, into the applicable NSSS systems and components evaluations, as well as safety analyses performed in support of the Return to RCS NOP/NOT Program.

2.1.2 Input Parameters and Assumptions

The NSSS design parameters provide the RCS and secondary-side system conditions (thermal power, temperatures, pressures, and flows) that are used as the basis for the design transients and for the systems, structures, components, accidents, and fuel analyses and evaluations.

The major input parameters considered for the four cases of NSSS design parameters provided in Table 2.1-1 support the Return to RCS NOP/NOT Program and include the following.

1. NSSS Power of 3327 MWt, which includes a reactor coolant pump (RCP) net heat input of 12 MWt

2. A nominal reactor coolant pressure of 2250 psia

3. Reactor vessel average temperature (T_{avg}) range of 553.7°F to 575.4°F

4. Feedwater temperature (T_{feed}) of 437.4°F

5. Babcock & Wilcox International (BWI) Series 51 replacement steam generators (RSGs)

6. Two core bypass flows:

   - Core bypass flow of 5.1percent, which accounts for intermediate flow mixing vanes (IFMs) and thimble plugs installed (TPI)

   - Core bypass flow of 7.1 percent, which accounts for IFMs and thimble plugs removed (TPR)

7. Westinghouse 15x15 Upgrade Fuel

8. SGTP levels of 0 and 30 percent

9. TDF of 83,200 gpm/loop

10. Minimum measured flow (MMF) of 339,100 gpm total

The major input parameters used in the calculation of the four cases of NSSS design parameters provided in Table 2.1-2 also support the Return to RCS NOP/NOT Program with reduced SGTP and increased TDF and include the following.

1. NSSS Power of 3327 MWt, which includes an RCP net heat input of 12 MWt

2. A nominal reactor coolant pressure of 2250 psia

3. T_{avg} range of 553.7°F to 575.4°F

4. T_{feed} of 437.4°F

5. BWI Series 51 RSGs

6. Two core bypass flows:
   - Core bypass flow of 5.1 percent, which accounts for IFMs and TPI
   - Core bypass flow of 7.1 percent, which accounts for IFMs and TPR

7. Westinghouse 15x15 Upgrade Fuel

8. SGTP levels of 0 and 10 percent

9. TDF of 88,500 gpm/loop

10. MMF of 362,900 gpm total

There were no major assumptions used for this evaluation.

2.1.3 Acceptance Criteria

The primary acceptance criteria for the NSSS design parameters dictate that the parameters must provide Cook Unit 1 adequate flexibility and margin for NSSS and secondary system operation, while bounding the expected operating conditions. Achievement of these acceptance criteria is assessed through the downstream NSSS evaluations and analyses that use the NSSS design parameters as inputs. These NSSS evaluations and analyses will be performed as part of the Cook Unit 1 Return to RCS NOP/NOT Program.

2.1.4 Description of Analyses and Evaluations

The computer code used to determine the NSSS design parameters was NSSSPlus. There is no explicit United States Nuclear Regulatory Commission (USNRC) approval for the code since it is used to facilitate calculations that could be performed by hand; however, NSSSPlus is under the Westinghouse Software Configuration Control. It uses basic thermal-hydraulic calculations, along with first principles of engineering, to generate the temperatures, pressures, and flows shown in Tables 2.1-1 and 2.1-2. The code and method used to calculate these values have been successfully used to license previous uprates and other analyses for Westinghouse plants.

Tables 2.1-1 and 2.1-2 provide the NSSS design parameter cases generated and used as the basis for the Cook Unit 1 Return to RCS NOP/NOT Program. The cases are defined as follows:

- Cases 1 and 2 of Table 2.1-1 and Cases 5 and 6 of Table 2.1-2 represent parameters based on T_{avg} of 553.7°F. Case 2 of Table 2.1-1 and Case 6 of Table 2.1-2, which are based on the higher SGTP level, yield the minimum secondary-side steam generator pressure and temperature.

- Cases 3 and 4 of Table 2.1-1 and Cases 7 and 8 of Table 2.1-2 represent parameters based on T_{avg} of 575.4°F. Case 3 of Table 2.1-1, and Case 7 of Table 2.1-2, which are based on an average 0 percent SGTP, yield the maximum secondary-side steam pressure and temperature.

2.1.5 Results

Tables 2.1-1 and 2.1-2 contain the eight cases of NSSS design parameters for the Cook Unit 1 Return to RCS NOP/NOT Program.

2.1.6 Conclusions

The eight cases of NSSS design parameters identified for Cook Unit 1 in Tables 2.1-1 and 2.1-2 were used by Westinghouse as the basis for the NSSS analytical efforts for the Return to RCS NOP/NOT Program. The appropriate NSSS design parameters were used by the Westinghouse engineers for each NSSS analysis, based on the conditions that are most limiting for that analytical area.

Table 2.1-1 NSSS Design Parameters – Cook Unit 1 – Return to RCS NOP/NOT Program

	Case 1	Case 2	Case 3	Case 4
NSSS Power, MWt	3327	3327	3327	3327
10^6 Btu/hr	11,352	11,352	11,352	11,352
Reactor Power, MWt	3315	3315	3315	3315
10^6 Btu/hr	11,311	11,311	11,311	11,311
Thermal Design Flow, gpm/loop	83,200	83,200	83,200	83,200
Reactor 10^6 lb/hr	130.1	130.1	126.5	126.5
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	7.1(1,2)	7.1(1,2)	7.1(1,3)	7.1(1,3)
Reactor Coolant Temperature, °F				
Core Outlet	593.1(2)	593.1(2)	613.6(3)	613.6(3)
Vessel Outlet	588.2	588.2	609.1	609.1
Core Average	557.6(2)	557.6(2)	579.5(3)	579.5(3)
Vessel Average	553.7	553.7	575.4	575.4
Vessel/Core Inlet	519.2	519.2	541.7	541.7
Steam Generator Outlet	518.9	518.9	541.5	541.5
Steam Generator				
Steam Outlet Temperature, °F	500.4	489.4	523.9(4)	513.1
Steam Outlet Pressure, psia	684	618	840(4)	765
Steam Outlet Flow, 10^6 lb/hr total	14.46	14.44	14.53(4)	14.50
Feed Temperature, °F	437.4	437.4	437.4	437.4
Steam Outlet Moisture, % max.	0.10	0.10	0.10	0.10
Tube Plugging Level, %	0	30	0	30
Zero Load Temperature, °F	547	547	547	547
Hydraulic Design Parameters				
Mechanical Design Flow, gpm/loop	99,700			
Minimum Measured Flow, gpm total	339,100			

Notes:
1. Core Bypass Flow accounts for IFMs and TPR.
2. If thimble plugs are installed, the core bypass flow is 5.1%, core outlet temperature is 591.7°F, and core average temperature is 556.8°F.
3. If thimble plugs are installed, the core bypass flow is 5.1%, core outlet temperature is 612.3°F, and core average temperature is 578.8°F.
4. If a high steam pressure is more limiting for analysis purposes, a greater steam pressure of 856 psia, steam temperature of 526.0°F, and total steam flow of 14.54x10^6 lb/hr should be assumed. This is to envelop the possibility that the plant could operate with better than expected steam generator performance.

Table 2.1-2 NSSS Design Parameters – Cook Unit 1 – Return to RCS NOP/NOT with Reduced SGTP & Increased TDF

	Case 5	Case 6	Case 7	Case 8
NSSS Power, MWt	3327	3327	3327	3327
10^6 Btu/hr	11,352	11,352	11,352	11,352
Reactor Power, MWt	3315	3315	3315	3315
10^6 Btu/hr	11,311	11,311	11,311	11,311
Thermal Design Flow, gpm/loop	88,500	88,500	88,500	88,500
Reactor 10^6 lb/hr	138.0	138.0	134.2	134.2
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	7.1(1,2)	7.1(1,2)	7.1(1,3)	7.1(1,3)
Reactor Coolant Temperature, °F				
Core Outlet	590.8(2)	590.8(2)	611.5(3)	611.5(3)
Vessel Outlet	586.3	586.3	607.2	607.2
Core Average	557.3(2)	557.3(2)	579.2(3)	579.2(3)
Vessel Average	553.7	553.7	575.4	575.4
Vessel/Core Inlet	521.1	521.1	543.6	543.6
Steam Generator Outlet	520.9	520.9	543.4	543.4
Steam Generator				
Steam Outlet Temperature, °F	501.9	499.1	525.3(4)	522.6
Steam Outlet Pressure, psia	693	675	851(4)	831
Steam Outlet Flow, 10^6 lb/hr total	14.46	14.46	14.54(4)	14.53
Feed Temperature, °F	437.4	437.4	437.4	437.4
Steam Outlet Moisture, % max.	0.10	0.10	0.10	0.10
Tube Plugging Level, %	0	10	0	10
Zero Load Temperature, °F	547	547	547	547
Hydraulic Design Parameters				
Mechanical Design Flow, gpm/loop	99,700			
Minimum Measured Flow, gpm total	362,900			

Notes:
1. Core Bypass Flow accounts for IFMs and TPR.
2. If thimble plugs are installed, the core bypass flow is 5.1%, core outlet temperature is 589.5°F, and core average temperature is 556.6°F.
3. If thimble plugs are installed, the core bypass flow is 5.1%, core outlet temperature is 610.2°F, and core average temperature is 578.5°F.
4. If a high steam pressure is more limiting for analysis purposes, a greater steam pressure of 870 psia, steam temperature of 527.9°F, and total steam flow of 14.55×10^6 lb/hr should be assumed. This is to envelop the possibility that the plant could operate with better than expected steam generator performance.

3 DESIGN TRANSIENTS

3.1 NSSS DESIGN TRANSIENTS

Nuclear steam supply system (NSSS) design transients were specified in the original design analyses of NSSS component cyclic behavior. The selected transients are conservative representations of transients that, when used as a basis for component fatigue analysis, provide confidence that the component is appropriate for its application over the plant operating license period. The reactor coolant system (RCS) and its components are designed to withstand the cyclic load effects from RCS temperature and pressure changes. The current design transients were evaluated for their continued applicability for the return to RCS Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) conditions at Cook Unit 1.

The NSSS design transients are based on NSSS design parameters such as RCS hot leg and cold leg temperatures, steam generator (SG) secondary side steam and feedwater temperatures, and RCS thermal design flow at full power and no-load conditions. The return of Cook Unit 1 to NOP/NOT conditions will revise the current plant operating conditions. No other key input parameters for the NSSS design transients are changing for the Return to RCS NOP/NOT Program.

An evaluation has been performed to evaluate the effect of the revised operating conditions on the current NSSS design transients. The current design transients support an NSSS power level up to 3600 MWt, a full power vessel average temperature (T_{avg}) window from 547°F to 581.3°F, operating pressure of either 2000 psia or 2250 psia, and steam generator tube plugging (SGTP) level of up to 30 percent. The NOP/NOT conditions include a lower NSSS power of 3512 MWt, a narrower full power T_{avg} window from 553.7°F to 575.4°F, a single operating pressure of 2250 psia, and maximum SGTP level of 30 percent. Therefore, the conditions supported by the current design transients bound the NOP/NOT conditions.

The evaluation concluded that the current NSSS design transients bound the NOP/NOT full power design conditions and no other key design transient input parameters are impacted by the Return to RCS NOP/NOT Program. Therefore, the current NSSS design transient responses remain conservative and valid for the return to NOP/NOT conditions at Cook Unit 1. The current number of occurrences for each of the NSSS design transients in Table 4.1-10 of the Updated Final Safety Analysis Report (UFSAR) remains valid for the 60 year licensed plant life. No new transients need to be added as part of the return to NOP/NOT conditions at Cook Unit 1. Consistent with Reference 1, the full power steam pressure will continue to be limited to a minimum of 679 psia (administratively limited to 690 psia for conservatism), so that the primary to secondary differential pressure limit is not exceeded for any normal or upset condition design transients. This limitation on the plant operating conditions is reflected in UFSAR Chapter 4.2.2.4.

3.1.1 References

1. Donald C. Cook Nuclear Plant, Unit 1 – Issuance of Amendment 273 Regarding Measurement Uncertainty Recapture Power Uprate (TAC NO. MB5498), December 20, 2002.

4 NSSS SYSTEMS

4.1 PLANT OPERABILITY, MARGIN-TO-TRIP, AND PRESSURE CONTROL COMPONENT SIZING

4.1.1 Introduction

The various control system analyses of record (AOR) were evaluated for their continued applicability for the return of Cook Unit 1 to reactor coolant system (RCS) normal operating pressure/normal operating temperature (NOP/NOT) conditions. These control system analyses include; control system stability, margin-to-trip, and pressure control component sizing for the following design basis operational transients. These transients are consistent with the control systems design basis as documented in Updated Final Safety Analysis Report (UFSAR) Chapter 7.3.1.

- Ramp increase in turbine load at 1%/min between 20 and 100 percent power (Condition I)
- Ramp decrease in turbine load at 5%/ min 100 and 20 percent power (Condition I)
- 10 percent decrease in turbine load at a maximum rate of 200%/min (Condition I)
- 40 percent load rejection at a maximum rate of 200%/min (Condition I)

The majority of the analyses was performed as part of the rerating and reduced temperature and pressure program, as summarized in Reference 1. The rerating power level was not implemented and the control system analyses were subsequently evaluated for applicability to a measurement uncertainty recapture (MUR) power uprate as summarized in Reference 2.

Additionally, a turbine trip without a reactor trip transient from the P-8 permissive setpoint was analyzed to demonstrate that the intent of NUREG-0737, item II.K.3.10 (Reference 3), is satisfied.

4.1.2 Input Parameters and Assumptions

The plant operability, margin-to-trip, and pressure control component sizing analyses are based on the nuclear steam supply system (NSSS) design parameters at full power and no-load conditions. The NSSS design parameters for the Cook Return to RCS NOP/NOT Program are specified in Section 2.1.

In addition to the NSSS design parameters, key inputs to the control system analyses include NSSS control systems logic, settings, and their capacities and performance capabilities, as well as protection system logic and setpoints. No changes are being made to the control system logic and setpoints besides the required changes to full-load programmed vessel average temperature (T_{avg}), nominal pressurizer pressure, and the pressurizer water level program. Additionally, the low pressurizer pressure reactor trip nominal setpoint will be increased from 1875 to 1950 psig as part of the Return to RCS NOP/NOT Program. This has been evaluated with respect to the margin-to-trip analyses. No other key input parameters to the control systems analyses are changing as a result of the Return to RCS NOP/NOT Program.

4.1.3 Acceptance Criteria

The plant operability and margin-to-trip analyses are performed to demonstrate adequate operating margin is maintained to the relevant reactor trip system (RTS) and engineered safety feature actuation system (ESFAS) setpoints during and following the Condition I (normal operating) transients. All control system responses should be smooth and stable without diverging oscillations.

The turbine trip without a reactor trip from the P-8 permissive transient is analyzed to show that adequate margin is maintained to the pressurizer power-operated relief valve (PORV) actuation setpoint. The results of this analysis are used to demonstrate that the intent of NUREG-0737, item II.K.3.10 (Reference 3), is satisfied.

4.1.4 Description of Analyses and Evaluations

The current control systems analyses for Cook Unit 1 were evaluated for the return to NOP/NOT conditions. The evaluation consisted of reviewing the various AORs for impact due to the changes in the normal operating pressure and temperature and an increase in the low pressurizer pressure reactor trip nominal setpoint. No other key input parameters to the analyses are changing for the Return to RCS NOP/NOT Program. No transient reanalysis or simulations were required for this evaluation.

4.1.5 Results

The analyses in Reference 1 support a NSSS power level up to 3600 MWt, a full power T_{avg} window from 547°F to 581.3°F, and operating pressure of either 2100 psia or 2250 psia. Compared to the Reference 1 analyses, the NOP/NOT conditions include a lower NSSS power of 3327 MWt (licensed core power, plus uncertainties, and reactor coolant pump heat), a reduced full power T_{avg} window from 553.7°F to 575.4°F, and a single operating pressure of 2250 psia. The analyses that were performed as part of the MUR, as summarized in Reference 2, and the turbine trip without a reactor trip from the P-8 permissive analyses were also evaluated. The evaluation determined that the results and conclusions of these analyses remain valid for the return of Cook Unit 1 to NOP/NOT conditions. Therefore, the control system analyses for Cook Unit 1 remain valid for the NOP/NOT conditions in Section 2.1.

As part of the Return to RCS NOP/NOT Program, the low pressurizer pressure reactor trip nominal setpoint is being increased from 1875 to 1950 psig. This change is intended to increase the margin to the low pressurizer pressure safety injection nominal setpoint of 1775 psig following a reactor trip on low pressurizer pressure, as well as make the Cook Unit 1 trip setpoint consistent with the Cook Unit 2 setpoint. The control systems evaluation concluded that the margin-to-trip that is lost due to the 75 psi increase in the low pressurizer pressure reactor trip nominal setpoint will be offset by the 150 psi increase in nominal pressurizer pressure for the Return to RCS NOP/NOT Program. Furthermore, the margin to the low pressurizer pressure reactor trip nominal setpoint was reviewed for the limiting operational transients and it was determined that acceptable margin will remain to the revised setpoint of 1950 psig. Therefore, the increased low pressurizer pressure reactor trip nominal setpoint is acceptable with respect to the margin to trip during the Condition I transients.

4.1.6 Conclusions

Based on the preceding evaluations, the current control system analyses remain valid for the return of Cook Unit 1 to NOP/NOT conditions. The conditions supported by each of the control system analyses for Cook Unit 1 remain valid for the NOP/NOT conditions and acceptable margin-to-trip will be maintained for an increase in the low pressurizer pressure reactor trip nominal setpoint to 1950 psig.

4.1.7 References

1. WCAP-11902, Supplement 1, "Rerated Power and Revised Temperature and Pressure Operation for Donald C. Cook Nuclear Plant Units 1 & 2 Licensing Report," September 1989.

2. Donald C. Cook Nuclear Plant, Unit 1 – Issuance of Amendment 273 Regarding Measurement Uncertainty Recapture Power Uprate (TAC NO. MB5498), December 20, 2002.

3. NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.K.3.10, Proposed Anticipatory Trip Modification, November 1980.

5 NSSS ACCIDENT ANALYSES

5.1 LOSS-OF-COOLANT ACCIDENT (LOCA) TRANSIENTS

5.1.1 Best-Estimate Large-Break LOCA

Cook Unit 1 received approval to implement the Automated Statistical Treatment of Uncertainty Method (ASTRUM) Evaluation Model (EM) (Reference 1) by Amendment No. 306 dated October 17, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082670351, Reference 2). Subsequently, by letter dated March 19, 2012 (ADAMS Accession No. ML12088A104, Reference 3), as supplemented by letter dated June 11, 2012 (ADAMS Accession No. ML12173A025, Reference 4), Indiana Michigan Power Company (IM) submitted a report describing the impact of fuel pellet thermal conductivity degradation (TCD) on the emergency core cooling system (ECCS) evaluation model, and an estimate of the effect on the predicted peak cladding temperature (PCT) for Cook Unit 1. This report was submitted pursuant to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Section 50.46, paragraph (a)(3), and referred to a letter from Westinghouse Electric Company dated March 7, 2012 (ADAMS Accession No. ML12072A035, Reference 5). By letter dated March 7, 2013 (ADAMS Accession No. ML13077A137, Reference 9), the United States Nuclear Regulatory Commission (USNRC) found the assessment reported in Reference 3 to be acceptable. The estimate of effect accounted for a removal of the allowable nominal full power vessel average temperature (T_{avg}) range that was established in Reference 2.

In order to support a Return to Reactor Coolant System (RCS) Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) Program, the effects of explicitly modeling fuel pellet TCD on the Cook Unit 1 best-estimate large-break loss-of-coolant accident (BE LBLOCA) analysis have been re-considered following the method described in Reference 5 at NOP/NOT conditions. Due to the non-linear effects of the design input changes (which were updated relative to the assessment reported in Reference 3), the return to NOP/NOT evaluation is being assessed against the Cook Unit 1 BE LBLOCA analysis of record (AOR), which was submitted in Reference 7 and approved by the USNRC in Reference 2. The return to NOP/NOT evaluation supersedes the assessment reported in Reference 3. Additionally, due to different cases becoming limiting at NOP/NOT conditions, the prior PCT assessment reported in Reference 10 is also re-considered (Table 5.1.1-5).

Fuel performance data that accounts for fuel pellet TCD was used as input to the Cook Unit 1 evaluation. The new plant-specific performance analysis for design (PAD) fuel performance data were generated using an unlicensed model that includes explicit modeling of fuel pellet TCD, as described in Reference 5, and found acceptable by the USNRC for estimating the effect on the PCT for Cook Unit 1 in Reference 9. Therefore, the evaluation considers the fuel pellet TCD effects cited in USNRC Information Notice 2011-21 (Reference 6).

A quantitative evaluation for Cook Unit 1, as discussed in Reference 5, was performed to assess the PCT effect of NOP/NOT design input changes, and the effect of fuel TCD and peaking factor burndown on the Cook Unit 1 LBLOCA analysis. The evaluation concluded that the estimated PCT impact of fuel TCD and peaking factor burndown is +404°F, and the estimated PCT impact of the design input changes is -489°F. The peaking factor burndown included in the evaluation is provided in Table 5.1.1-1.

The nuclear design (ND) proposed peaking factor limits at beginning of cycle (BOC) for the Return to RCS NOP/NOT Program are compared with those from the AOR in Table 5.1.1-2. TCD and design input changes were evaluated with these proposed limits and it was demonstrated that the PCT results remained below the 10 CFR 50.46(b)(1) acceptance criterion. The treatment of the peaking factor uncertainties in the evaluation was handled consistent with the ASTRUM EM (Reference 1). The rationale for lowering the peaking factor limits was to reduce the calculated PCT through a reduction in local peaking and linear heat rates. Subsequent administrative controls as part of the implementation of the Return to RCS NOP/NOT Program include updates to the Cook Unit 1 Core Operating Limit Report (COLR).

To demonstrate compliance with the 10 CFR 50.46(b)(1) acceptance criterion concerning PCT when explicitly considering fuel pellet TCD and peaking factor burndown in the Cook Unit 1 LBLOCA analysis, design input values were revised to provide offsetting margins for the Return to RCS NOP/NOT Program (Table 5.1.1-3). These input changes are not changes to the approved 2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM.

In order to recapture margins while still maintaining a conservatively low containment backpressure, the LOTIC2 containment pressure response was calculated including the design input changes. The design input changes have the effect of increasing the LOTIC2 calculated backpressure. The final WCOBRA/TRAC (WC/T) input used in the return to NOP/NOT evaluation was compared to the updated LOTIC2 calculated pressure, consistent with Westinghouse BE LBLOCA analysis guidance, and is conservatively low (as required per Sections 11-3-1, 11-4-11, and 12-3-4 of Reference 1). This comparison is shown in Figure 5.1.1-1.

IM and Westinghouse utilize processes which ensure that LOCA analysis input values conservatively bound the as-operated plant values for those parameters. This is confirmed via the fuel reload process, the purpose of which is to evaluate plant changes resulting from the loading of different or new fuel into the core. As described in Reference 8, the evaluations performed for the fuel reload are reviewed in accordance with 10 CFR 50.59. The accident AORs are generally analyzed such that the relevant parameters are selected in a bounding direction compared to the expected operational values. In accordance with Reference 8, the fuel reload evaluation relies upon the bounding approach in which safety analyses are performed to accommodate the plant changes resulting from different or new fuel in the core without requiring new safety analyses.

As part of the reload evaluation, the LOCA analyst generates a list of important parameters to the LBLOCA analysis which shows a fuel reload dependency, and identifies the values of those parameters supported by the LBLOCA licensing basis analyses and evaluations. The parameters are confirmed to support the reload nuclear design or are evaluated with respect to the LBLOCA analysis.

Separate from the fuel reload process, plant changes which may impact the LBLOCA analysis are identified to Westinghouse, and 10 CFR 50.46 evaluations are performed as necessary. During the reload process, a summary of plant changes implemented since the previous cycle and changes planned for the upcoming cycle is provided by IM to Westinghouse. Westinghouse reviews those changes identified by IM to ensure the non-reload related parameters analyzed in the LBLOCA analysis remain applicable. For example, steam generator tube plugging (SGTP) level is one such non-reload related parameter reviewed as part of the reload analysis to demonstrate that the LBLOCA analysis remains applicable.

The return to NOP/NOT evaluation included an improved timestep interval of data transmittal from WC/T to HOTSPOT. This was included to more accurately model phenomena such as decay heat generation in HOTSPOT. The reduction of the timestep interval is considered an improvement to the internal processing of data and is not a change to the approved 2004 ASTRUM EM.

HOTSPOT Version 8.0 was used for the return to NOP/NOT evaluation. The error corrections, code improvements, and miscellaneous code cleanup between the HOTSPOT code version used in the AOR (Version 6.1) versus the return to NOP/NOT evaluation (Version 8.0) are described in the 10 CFR 50.46 reporting (previously issued to the USNRC) provided in Table 5.1.1-6. The error corrections and code improvements between HOTSPOT versions do not impact the thermal conductivity model, and have an inconsequential impact on the results of the evaluation.

The evaluation method discussed in Reference 5 was used to determine the estimated effect of fuel pellet TCD and peaking factor burndown. First, the Integrated PCT, which considers design input changes and TCD and peaking factor burndown, was calculated to demonstrate compliance with the 10 CFR 50.46(b)(1) criterion. Then, the Margin Recovery PCT, which only considers design input changes, was calculated.

The evaluation considers a range of rod burnup values over the life of the fuel to determine the impact of fuel TCD and related burnup effects. A total of 45 WC/T executions (referred to as "cases" hereafter) were performed to determine the Integrated PCT value for the Cook Unit 1 return to NOP/NOT evaluation. The cases were selected based on the AOR PCT and the range of rod burnup values where TCD affects fuel rod conditions. All AOR cases with a PCT over 1600°F were explicitly re-executed for the Integrated PCT calculation; this includes the top 25 percent of the original 124-runset ASTRUM analysis (31 cases). The hot rod burnup for each case was maintained, translated, or both to cover the burnup range of interest. All other AOR uncertainty attributes were maintained. HOTSPOT executions were performed for each WC/T case to consider the effect of local uncertainties for both IFBA (integral fuel burnable absorber) and non-IFBA fuel. The most limiting PCT result was selected as the Integrated PCT value.

For the return to NOP/NOT evaluation, all cases selected for inclusion in the Integrated PCT calculation were also selected for the margin recovery runset. This includes the top 31 AOR cases (top 25 percent of all AOR cases). The most limiting result was selected as the Margin Recovery PCT to allow estimation of the effect of the margins taken on PCT.

In this evaluation, engineering judgment was applied to select a run set of limiting cases for the purpose of evaluating the effects of the design input margins and TCD on the Cook Unit 1 LBLOCA PCT. It is noted that the uncertainty attributes for the selected cases were maintained from the original 124-run ASTRUM analysis. The remaining cases from the ASTRUM AOR which were not explicitly evaluated are expected to remain non-limiting and therefore would not affect the PCT estimate. The return to NOP/NOT evaluation supports the full life of the fuel operation.

The estimate of effect for the design input changes was provided by the difference between the AOR PCT and the Margin Recovery PCT. The estimate of effect of TCD at NOP/NOT conditions was provided by the difference between the Integrated PCT and the Margin Recovery PCT. The limiting Integrated PCT case, considering all design input changes and TCD and peaking factor burndown, was 2043°F, which is less than the 2200°F PCT acceptance criterion. Considering only the design input changes, the Margin Recovery PCT was 1639°F.

Given the AOR PCT (2128°F), the estimated effect of the design input changes for 10 CFR 50.46 reporting purposes is -489°F. The estimated effect of TCD and peaking factor burndown is the difference between the Margin Recovery PCT and the Integrated PCT, or +404°F. The results of the evaluation are summarized in Table 5.1.1-4. It is noted that the estimated effect of TCD and peaking factor burndown is within 25°F of the original TCD estimate.

5.1.1.1 References

1. Westinghouse Report WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment Of Uncertainty Method (ASTRUM)," January 2005. (Westinghouse Proprietary Class 2)

2. Letter from Terry A. Beltz (USNRC) to Michael W. Rencheck (AEP), "Donald C. Cook Nuclear Plant, Unit 1 – Issuance of Amendment to Renewed Facility Operating License Regarding Use of the Westinghouse ASTRUM Large Break Loss-of-Coolant Accident Analysis Methodology (TAC NO. MD7556)," dated October 17, 2008. (Available in USNRC ADAMS under Accession Number ML082670351.)

3. Letter from Joel P. Gebbie (AEP) to the USNRC, "Donald C. Cook Nuclear Plant Units 1 and 2 Response to Information Request Pursuant to 10 CFR 50.54(f) Related to the Estimated Effect on Peak Cladding Temperature Resulting from Thermal Conductivity Degradation in the Westinghouse-Furnished Realistic Emergency Core Cooling System Evaluation (TAC NO. M99899)," dated March 19, 2012. (Available in USNRC ADAMS under Accession Number ML12088A104.)

4. Letter from Michael H. Carlson (AEP) to the USNRC, "Donald C. Cook Nuclear Plant Units 1 and 2 Response to Request for Information 10 CFR 50.46 Report for Emergency Core Cooling System Model Change or Error Associated with Thermal Conductivity Degradation," dated June 11, 2012. (Available in USNRC ADAMS under Accession Number ML12173A025.)

5. Letter from James A. Gresham (Westinghouse) to the USNRC, "Westinghouse Input Supporting Licensee Response to NRC 10 CFR 50.54(f) Letter Regarding Nuclear Fuel Thermal Conductivity Degradation (Proprietary/Non-Proprietary)," dated March 7, 2012. (Available in USNRC ADAMS under Accession Number ML12072A035.)

6. USNRC Information Notice 2011-21 from Timothy J. McGinty and Laura A. Dudes, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation," dated December 13, 2011. (Available in USNRC ADAMS under Accession Number ML113430785.)

7. Letter from Joseph N. Jensen (AEP) to the USNRC, "Donald C. Cook Nuclear Plant Unit 1, Docket No. 50-315, License Amendment Request Regarding Large Break Loss-of-Coolant Accident Analysis Methodology," dated December 27, 2007. (Available in USNRC ADAMS under Accession Number ML080090268.)

8. Westinghouse Report WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985. (Westinghouse Proprietary Class 2)

9. Letter from Thomas J. Wengert (USNRC) to Lawrence J. Weber (AEP), "Donald C. Cook Nuclear Plant, Units 1 and 2 – Evaluation of Report Concerning Significant Emergency Core Cooling System Evaluation Model Error Related to Nuclear Fuel Thermal Conductivity Degradation (TAC NOS. ME8322 and ME8323)," dated March 7, 2013. (Available in USNRC ADAMS under Accession Number ML13077A137.)

10. Letter from Joel P. Gebbie (AEP-NRC-2013-68) to USNRC, "Donald C. Cook Nuclear Plant Unit 1, Docket No. 50-315, 30-Day Report of Changes to OR Errors in an Evaluation Model," dated August 30, 2013.

Table 5.1.1-1 Cook Unit 1 Peaking Factors versus Rod Burnup

Rod Burnup (MWD/MTU)	Steady State FQ	Transient FQ[1]	$F_{\Delta H}$[1,2]
0	1.650	2.090	1.530
28,000	1.650	2.090	1.530
30,000	1.460	1.850	1.391
60,000	1.360	1.720	1.268
62,000	1.360	1.720	1.268

Notes:
1. Includes uncertainties.
2. Hot assembly radial peaking factor uses same peaking factor burndown since it is a function of $F_{\Delta H}$.

Table 5.1.1-2 Cook Unit 1 Core Operating Peaking Factors for the AOR and the Return to RCS NOP/NOT Evaluation[1]

Peaking Factor	AOR[2]	Return to NOP/NOT
Transient FQ	2.150	2.090
$F_{\Delta H}$	1.550	1.530

Notes:
1. Includes uncertainties.
2. Values from Table 1 of Reference 7, which was approved by the USNRC in Reference 2.

Table 5.1.1-3 Cook Unit 1 Updated Design Inputs for the BE LBLOCA Return to RCS NOP/NOT Evaluation Considering the Effects of Fuel TCD		
Design Input	**Updated Value**	**AOR Value**
Hot Full Power Nominal T_{avg}[1,4]	571°F	$553.7°F \leq T_{avg} \leq 575.4°F$
Hot Full Power RCS Pressure[4]	2250 psia	2100 psia 2250 psia
Peaking Factors[5]	See Table 5.1.1-1	See Table 5.1.1-2
Conservatively Low Containment Pressure[5]	See Figure 5.1.1-1	See Figure 17 of Reference 7
Maximum Containment Spray Flow[5]	3600 gpm/pump	3700 gpm/pump
Minimum Containment Spray Flow Actuation Delay Time[5]	244 seconds	44 seconds
Minimum Containment Air Recirculation Fan Delay Time[5]	270 seconds	108 seconds
Accumulator (ACC) Temperature[6]	$70°F \leq T_{ACC} \leq 100°F$	$60°F \leq T_{ACC} \leq 120°F$
Safety Injection (SI) Flow[2,6]	Minimum	Minimum
SI Temperature[6]	$70°F \leq T_{SI} \leq 100°F$	$70°F \leq T_{SI} \leq 105°F$
SI Initiation Delay Time[3,6]	≤ 17 seconds (with offsite power) ≤ 28 seconds (without offsite power)	≤ 27 seconds (with offsite power) ≤ 54 seconds (without offsite power)

Notes:
1. Uncertainty band applied to nominal value remains unchanged from AOR.
2. Updated based on BE LBLOCA representative minimum refueling water storage tank (RWST) level and containment spilling assumption.
3. Updated based on current surveillance test requirements.
4. Proposed modification to normal operating pressure and normal operating temperature.
5. These updated design inputs provide the offsetting margins to reduce the calculated PCT for the Return to RCS NOP/NOT evaluation and were updated from the original TCD evaluation reported in Reference 3.
6. These updated design inputs were initially incorporated in the original TCD evaluation reported in Reference 3 and are carried forward to the return to RCS NOP/NOT evaluation.

Table 5.1.1-4 Summary of Cook Unit 1 Return to RCS NOP/NOT Evaluation				
AOR PCT (°F)	**Integrated PCT (°F)**	**Margin Recovery PCT (°F)**	**TCD/Peaking Factor Burndown PCT Penalty (°F)**	**Margin Update PCT Benefit (°F)**
2128	2043	1639	404	-489

Table 5.1.1-5 Summary of Cook Unit 1 PCT Assessments	
AOR PCT (°F)	2128
Margin Recovery PCT (°F)	1639
Integrated PCT (°F)	2043
Revised Overall PCT (°F)	1952[1]
Notes: 1. The Revised Overall PCT is the sum of the estimated effect of revised heat transfer multiplier (HTM) distributions at NOP/NOT conditions and the Integrated PCT.	

Table 5.1.1-6 Summary of Cook Unit 1 10 CFR 50.46 Reporting

GENERAL CODE MAINTENANCE – Discretionary Change (Reference 1)

Background

A number of coding changes were made as part of normal code maintenance. Examples include additional information in code outputs, improved automation and diagnostics in the codes, increased code dimensions, and general code cleanup. All of these changes are considered to be Discretionary changes in accordance with subsection 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to Pressurized Water Reactors (PWRs) with Upper Plenum Injection

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

The nature of these changes leads to an estimated PCT impact of 0°F for 10 CFR 50.46 reporting purposes.

Reference

1. Westinghouse Letter – LTR-NRC-09-17, Rev. 1, "U. S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2008," April 7, 2011.

HOTSPOT BURST TEMPERATURE LOGIC ERRORS – Non-Discretionary Change (Reference 1)

Background

The HOTSPOT code has been updated to incorporate the following corrections to the burst temperature logic: (1) change the rod internal pressure used to calculate the cladding engineering hoop stress from the value in the previous time step to the value in the current time step; (2) revise the average cladding heat-up rate calculation to reset selected variables to zero at the beginning of each trial and use the instantaneous heat-up rate when fewer than five values are available; and, (3) reflect the assumed saturation of ramp rate effects above 28°C/s for Zircaloy-4 cladding from Equation 7-66 of Reference 2. These changes represent a closely-related group of Non-Discretionary Changes in accordance with subsection 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

Sample calculations for each change showed no effect on peak cladding temperature, leading to an estimated impact of 0°F for 10 CFR 50.46 reporting purposes.

References

1. Westinghouse Letter – LTR-NRC-09-17, Rev. 1, "U. S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2008," April 7, 2011.
2. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2-5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," S. M. Bajorek et al., March 1998.

Table 5.1.1-6 Summary of Cook Unit 1 10 CFR 50.46 Reporting (cont.)

GENERAL CODE MAINTENANCE – Discretionary Change (Reference 1)

Background

Various changes have been made to enhance the usability of codes and to streamline future analyses. Examples of these changes include modifying input variable definitions, units and defaults; improving the input diagnostic checks; enhancing the code output; optimizing active coding; and eliminating inactive coding. These changes represent Discretionary Changes that will be implemented on a forward-fit basis in accordance with subsection 4.1.1 of WCAP-13451.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

The nature of these changes leads to an estimated PCT impact of 0°F.

Reference

1. Westinghouse Letter – LTR-NRC-10-75, "U. S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2009," January 10, 2011.

HOTSPOT GAP HEAT TRANSFER LOGIC – Non-Discretionary Change (Reference 1)

Background

The HOTSPOT code has been updated to incorporate the following changes to the gap heat transfer logic: (1) change the gap temperature from the pellet average temperature to the average of the pellet outer surface and cladding inner surface temperatures; (2) correct the calculation of the pellet surface emissivity to use a temperature in °R (as specified in Equation 7-28 of Reference 2) instead of °F; and (3) revise the calculation of the gap radiation heat transfer coefficient to delete a term and temperature adder not shown in or suggested by Equation 7-28 of Reference 2. These changes represent a closely related group of Non-Discretionary Changes in accordance with subsection 4.1.2 of WCAP-13451.

Affected Evaluation Model(s)

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

2004 Westinghouse Realistic Large Break LOCA Evaluation Model Using ASTRUM

Estimated Effect

Sample calculations showed a minimal impact on PCT, leading to an estimated effect of 0°F.

References

1. Westinghouse Letter – LTR-NRC-10-75, "U. S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2009," January 10, 2011.
2. WCAP-12945-P-A, Volume 1, Revision 2, "Code Qualification Document for Best Estimate LOCA Analysis, Volume I: Models and Correlations," March 1998.

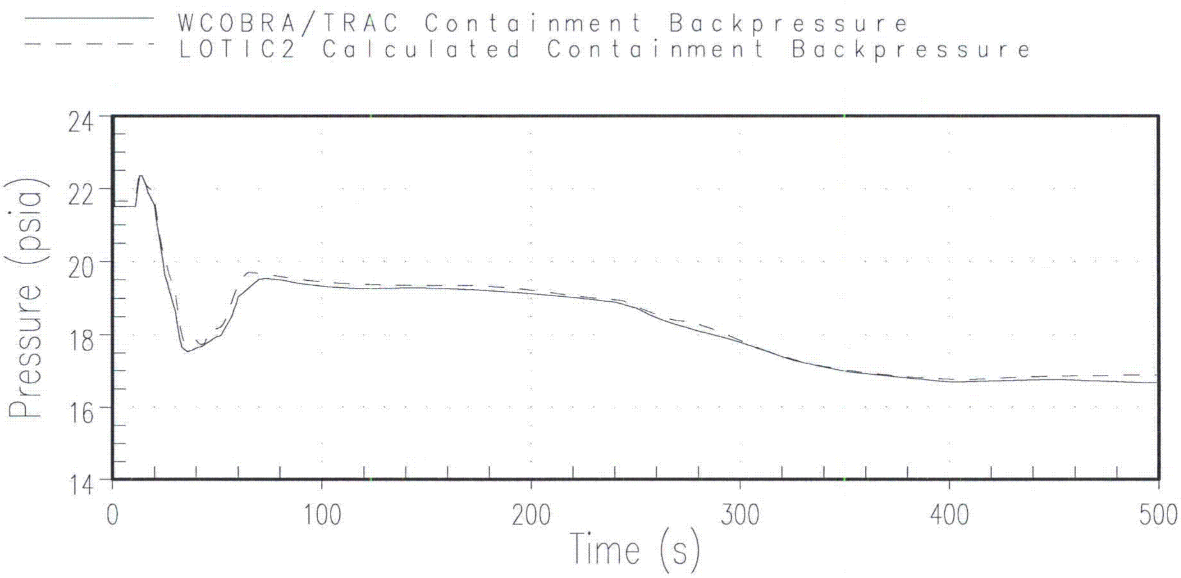


Figure 5.1.1-1 Analyzed Versus Calculated Containment Backpressure

5.1.2 Small-Break LOCA

5.1.2.1 Introduction and Background

A small-break loss-of-coolant accident (SBLOCA) evaluation was performed to assess the impact of the Cook Unit 1 Return to Reactor Coolant System (RCS) Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) Program on the SBLOCA analysis of record (AOR) documented in Reference 1. Additionally, a SBLOCA evaluation was performed against the input changes and margin sources associated with the best-estimate LOCA (BELOCA) evaluation (subsection 5.1.1). The purpose of the SBLOCA evaluation is to assess continued compliance with the 10 CFR 50.46 requirements at NOP/NOT conditions.

5.1.2.2 Input Parameters and Assumptions

The following key parameters were reviewed as part of the effort discussed herein:

1. Return to NOP – Nominal pressurizer pressure of 2250 psia

2. Return to NOT – Nominal vessel average temperature (T_{avg}) of 571.0°F

3. Revised nuclear steam supply system (NSSS) parameters (Section 2.1)

4. Sources of margin utilized in the BELOCA evaluation:

   - Increased containment spray (CTS) actuation delay time
   - Reduced maximum CTS system flow rate
   - Increased containment air recirculation fan delay time
   - Consideration of fuel peaking factor burndown limits

5. Increased low pressurizer pressure reactor trip nominal setpoint

6. Revised pressurizer water level program

5.1.2.3 Acceptance Criteria

The acceptance criteria for SBLOCA are those outlined by 10 CFR 50.46, namely:

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

2. Maximum cladding oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

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 the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

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

5. Long-term cooling. After any calculated successful initiation operation of the emergency core cooling system (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.

5.1.2.4 Description of Evaluation

A comparison was performed between the Cook Unit 1 SBLOCA AOR and the proposed changes associated with the Return to RCS NOP/NOT Program. Each change identified in subsection 5.1.2.2 was reviewed for impact against the inputs, assumptions and methodology utilized in the SBLOCA AOR.

5.1.2.5 Results

The SBLOCA AOR supports operation at nominal pressurizer pressures of 2100 and 2250 psia; therefore, the proposed return to an NOP of 2250 psia is explicitly supported by the current SBLOCA analysis documented in Reference 1. The AOR supports a nominal full-power T_{avg} range of 553.7°F to 575.4°F; therefore, the return to a full power nominal T_{avg} of 571.0°F is explicitly supported by the AOR. The revised NSSS design parameters generated to support the Return to RCS NOP/NOT Program, as discussed in Section 2.1, were reviewed and are consistent with the NSSS design parameters utilized in the SBLOCA AOR calculations. Each proposed source of margin identified in support of the BELOCA evaluation was confirmed as either not impacting the SBLOCA AOR or representing additional sources of margin relative to the SBLOCA AOR calculations. Plant specific pressurizer water level control programs are not modeled in the SBLOCA methodology. Additionally, the increased low pressurizer pressure reactor trip nominal setpoint is conservatively bounded by the setpoint analyzed in the SBLOCA AOR.

5.1.2.6 Conclusions

It is concluded that the NOP/NOT conditions have no impact on the SBLOCA AOR documented in Reference 1. The margin recovery actions being pursued to support the BELOCA evaluation have been confirmed to not adversely impact the SBLOCA AOR. The Cook Unit 1 AOR remains applicable and continues to meet the 10 CFR 50.46 requirements, as described in subsection 5.1.2.3, when considering the proposed changes to NOP and NOT.

5.1.2.7 References

1. AEP Letter, AEP-NRC-2012-71, "Donald C. Cook Nuclear Plant Unit 1, Revised Small Break Loss-of-Coolant Accident Analysis," August 2012 (ADAMS Accession Number ML12256A685).

5.1.3 Post-LOCA Long Term Cooling Subcriticality, Boric Acid Precipitation Control and Decay Heat Removal

5.1.3.1 Post-LOCA Long-Term Subcriticality

5.1.3.1.1 Introduction and Background

Post-loss-of-coolant accident (post-LOCA) long-term subcriticality analyses support evaluations that demonstrate the core will remain subcritical upon entering the sump recirculation phase of emergency core cooling system (ECCS) operation. During the sump recirculation phase, safety injection (SI) flow is drawn from the containment sump following switchover from the refueling water storage tank (RWST). Post-LOCA subcriticality analyses determine a sump mixed-mean boron concentration that accounts for the various water and boron contributors to the sump. The sump mixed-mean boron concentration calculations are used to develop a post-LOCA subcriticality boron limit curve that becomes a Reload Safety Analysis Checklist (RSAC) limit that is confirmed on a cycle-specific basis as part of the Westinghouse Reload Safety Evaluation (RSE) methodology (Reference 1). Note that for cold leg breaks, D. C.Cook credits the negative reactivity provided by insertion of the rod control cluster assemblies (RCCAs) to offset the post-LOCA dilution of boron in the containment sump due to the boron concentrating in the reactor vessel (Reference 2).

5.1.3.1.2 Input Parameters and Assumptions

The sump boron concentration model used for D. C. Cook's post-LOCA subcriticality AOR is based on the following:

1. The calculation of the sump mixed-mean boron concentration assumes minimum mass and minimum boron concentrations for significant boron sources such as the RWST, and maximum mass and minimum boron concentrations for significant dilution sources such as the reactor coolant system (RCS) and ice melt liquid.

2. Boron is uniformly distributed in the sump liquid. The post-LOCA sump inventory is made up of constituents that are equally likely to return to the containment sump; selective holdup in containment is neglected.

3. The sump mixed-mean boron concentration is calculated as a function of the pre-trip RCS conditions.

5.1.3.1.3 Acceptance Criteria

There are no specific acceptance criteria when calculating the post-LOCA sump mixed-mean boron concentration. The resulting sump boron concentration, which is calculated as a function of the pre-LOCA RCS boron concentration, becomes an RSAC limit that is reviewed for each cycle-specific core design to confirm that adequate boron exists to maintain subcriticality for the long-term time period following the switchover to cold leg recirculation.

5.1.3.1.4 Description of Analyses and Evaluations

D. C. Cook's post-LOCA subcriticality calculations were reviewed to determine whether the design inputs, assumptions, and methodology used therein support operation of Cook Unit 1 at the normal operating pressure (NOP) and normal operating temperature (NOT) of 2250 psia and 571°F, respectively.

5.1.3.1.5 Results

The review of D. C. Cook's post-LOCA subcriticality calculations concluded that the design inputs, assumptions, and methodology used therein support operation of Cook Unit 1 at NOP/NOT. In particular, the review concluded that the RCS liquid mass, ice melt rates, and total ice melt mass used in the calculations support operation at NOP/NOT.

5.1.3.1.6 Conclusions

The D. C. Cook post-LOCA subcriticality AOR supports operation of Cook Unit 1 at the NOP/NOT conditions of 2250 psia and 571°F, respectively.

5.1.3.2 Post-LOCA Long-Term Boric Acid Precipitation Control and Decay Heat Removal

5.1.3.2.1 Introduction and Background

Post-LOCA long-term cooling (LTC) analyses determine the time at which ECCS recirculation should be realigned to the RCS hot legs to prevent potential boron precipitation from occurring in the reactor vessel. These analyses are performed to satisfy the requirements of 10 CFR 50.46(b), Items 4 and 5:

(4) Coolable Geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(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.

The Westinghouse methodology used to demonstrate compliance with the requirements of 10 CFR 50.46(b) up to hot leg recirculation is documented in WCAP-8339 (Reference 3).

5.1.3.2.2 Input Parameters and Assumptions

The post-LOCA LTC AOR for Cook Unit 1 is the same as the post-LOCA LTC AOR for Cook Unit 2. A summary of D. C. Cook's boric acid precipitation control (BAPC) AOR was provided to the United States Nuclear Regulatory Commission (USNRC) in fulfillment of an American Electric Power (AEP) commitment to perform an updated analysis of the potential for boric acid precipitation to occur during the recirculation phase of a postulated LOCA (Reference 4).

The inputs used to perform post-LOCA LTC analyses include core power levels, fuel dimensions, and RCS and ECCS volumes, temperatures, pressures, and boron concentrations. For ice condenser plants such as Cook Unit 1, post-LOCA LTC analyses also include consideration for additional sump inventory due to ice melt.

The model used for D. C. Cook's post-LOCA BAPC AOR is based on the following assumptions and meets USNRC guidance as presented in Reference 5; it is also consistent with the interim methodology described in Reference 6.

1. The calculation of the sump mixed-mean boron concentration assumes maximum mass and maximum boron concentrations for significant boron sources such as the RWST liquid, and minimum mass and maximum boron concentrations for significant dilution sources such as the RCS and ice melt liquid.

2. Boron is uniformly distributed in the sump liquid. The post-LOCA sump inventory is made up of constituents that are equally likely to return to the containment sump; selective holdup in containment is neglected.

3. The sump mixed-mean boron concentration is calculated as a function of the pre-trip RCS conditions.

4. The core mixing volume accounts for voiding in the active fuel and upper plenum regions.

5. The core mixing volume includes 50 percent of the lower plenum as supported in Reference 6.

6. The core mixing volume considers the potential negative effects of loop pressure drop.

7. The boric acid incipient precipitation limit is the experimentally determined solubility limit of 29.27 weight percent reported in Reference 7.

8. The decay heat generation rate is based on 1971 American Nuclear Society (ANS) Standard for an infinite operating time with 20 percent uncertainty. The initial core power includes instrument uncertainty as required by Section 1.A of 10 CFR 50, Appendix K.

9. ECCS recirculation flows are shown to dilute the core and replace core boil-off, thus keeping the core quenched.

5.1.3.2.3 Acceptance Criteria

The acceptance criteria for post-LOCA LTC analyses are demonstrated by calculating a time by which ECCS flow to the RCS hot legs must be established in order to prevent precipitation of boric acid in the reactor vessel region. The time to establish ECCS flow to the RCS hot legs is determined using methods, plant design assumptions, and operating parameters that are consistent with the interim methodology reported in Reference 6.

5.1.3.2.4 Description of Analyses and Evaluations

D. C. Cook's post-LOCA LTC calculations were reviewed to ensure that the design inputs, assumptions, and methodology used therein support operation of Cook Unit 1 at the NOP/NOT conditions of 2250 psia and 571°F, respectively.

5.1.3.2.5 Results

The review of D. C. Cook's post-LOCA LTC calculations concluded that the design inputs, assumptions, and methodology used therein support operation of Cook Unit 1 at NOP/NOT. In particular, the review concluded that the RCS liquid mass, ice melt rates, ice melt mass, and ECCS flows used in the calculations support operation at NOP/NOT.

5.1.3.2.6 Conclusions

The D. C. Cook post-LOCA LTC AOR supports operation of Cook Unit 1 at the NOP/NOT conditions of 2250 psia and 571°F, respectively.

5.1.3.3 References

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

2. John F. Stang (USNRC) to Robert P. Powers (AEP), "Issuance of Amendments – Donald C. Cook Nuclear Plants, Units 1 and 2 (TAC Nos. MA6473 and MB6474)," December 23, 1999. (ML003672677)

3. WCAP-8339 (Non-Proprietary), "Westinghouse Emergency Core Cooling System Evaluation Model – Summary," June 1974.

4. M. H. Carlson (AEP) to USNRC Document Control Desk, "Donald C. Cook Nuclear Power Plant Unit 2 Docket No. 50-316 Updated Boric Acid Precipitation Analysis for Recirculation Phase of a Postulated Large-Break Loss-Of-Coolant Accident (TAC No. ME1017)," June 30, 2011. (ML11193A047)

5. S. E. Peters (USNRC) to S. L. Rosenberg (USNRC), "Summary of August 23, 2006 Meeting with the Pressurized Water Reactor Owners Group (PWROG) to Discuss the Status of Program to Establish Consistent Criteria for Post Loss-of-Coolant (LOCA) Calculations," October 3, 2006. (ML062690017)

6. Beaver Valley Extended Power Uprate Safety Evaluation Report, "Safety Evaluation Related to Extended Power Uprate at Beaver Valley Power Station, Unit Nos. 1 and 2," July 19, 2006. (ML061720376)

7. P. Cohen, Water Coolant Technology of Power Reactors, Chapter 6, "Chemical Shim Control and pH Effect," ANS-USEC Monograph, 1980 (Originally published in 1969).

5.1.4 LOCA Hydraulic Forces

5.1.4.1 Introduction and Background

Loss-of-coolant accident (LOCA) hydraulic forces are generated as input to the analyses of the reactor coolant system (RCS) components. The analyses of the components are performed in order to comply with Title 10 of the Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, General Design Criteria (GDC) 4 – Environmental and Dynamic Effects Design Bases.

> "Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluid that may result from equipment failures and events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

The dynamic effects design bases for Cook Unit 1 exclude the postulation of main loop pipe ruptures (Reference 1). Also, the pressurizer surge line on Cook Unit 1 is excluded from GDC-4 via leak-before-break (Reference 2).

5.1.4.2 Input Parameters and Assumptions

To conservatively calculate LOCA hydraulic forces for the Cook Unit 1 Return to Reactor Coolant System (RCS) Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) Program, the following operating conditions were considered in establishing the limiting temperature and pressures:

1. Initial RCS conditions associated with a minimum thermal design flow (TDF) of 83,200 gpm per loop

2. Reactor core power of 3315 MWt

3. A nominal RCS hot full power (HFP) T_{avg} of 553.7°F. This provides an RCS cold leg temperature (T_{cold}) of 519.2°F and an RCS hot leg temperature (T_{hot}) of 588.2°F.

4. An RCS temperature uncertainty of -4.1°F

5. A nominal RCS pressure of 2250 psia

6. A pressurizer pressure uncertainty of ±67 psia

Based on these conditions, the LOCA hydraulic forces were evaluated against a minimum T_{cold} of 515.1°F, a minimum T_{hot} of 584.1°F, and a pressurizer pressure of 2317 psia, including uncertainties.

5.1.4.3 Acceptance Criteria

There are no specific acceptance criteria. The results of this work (LOCA forces) are used as input to support a return to NOP/NOT for Cook Unit 1. The structural analyses performed using these forcing functions were performed to demonstrate compliance with 10 CFR 50, Appendix A, GDC-4.

5.1.4.4 Description of Analyses and Evaluations

LOCA forces analyses are performed to support different system and component analyses. Reactor vessel LOCA forces analyses support qualification of the reactor vessel, reactor vessel supports, and reactor vessel internals, including fuel qualification. Reactor coolant loop LOCA forces analyses support qualification of the reactor coolant loop piping and associated piping supports. These analyses were performed using different models with a focus on the component of interest.

All LOCA hydraulic forces analyses for the Cook Unit 1 Return to RCS NOP/NOT Program were performed directly at the analyzed reactor power level of 3315 MWt using models specific to the Cook Unit 1 nuclear steam supply system (NSSS) design. The results of the LOCA hydraulic forces analyses were then used as input to the calculations for component qualification.

5.1.4.5 Results

Existing LOCA forces analyses, which support the qualification of Cook Unit 1, have been evaluated for Unit 1 NOP/NOT conditions. LOCA hydraulic forces increase with lower temperatures and higher pressures. For Cook Unit 1, the currently supported operating conditions for LOCA hydraulic forces on the vessel/internals and loop bound the proposed conditions for the Return to RCS NOP/NOT Program.

5.1.4.6 Conclusions

The current LOCA forces analyses that support the qualification of Cook Unit 1 reactor pressure vessel, vessel internals, fuel, and loop piping were determined to remain bounding and applicable at the NOP/NOT conditions.

5.1.4.7 References

1. Westinghouse Report WCAP-15132, Revision 1 (Non-Proprietary), "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the D. C. Cook Units 1 and 2 Nuclear Power Plants," October 1999.

2. Westinghouse Report WCAP-15435, Revision 1 (Non-Proprietary), "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for D. C. Cook Units 1 and 2 Nuclear Power Plants," August 2000.

5.2 NON-LOCA TRANSIENTS

5.2.1 Introduction and Background

Chapter 14, "Safety Analysis [Unit 1]," of the D. C. Cook Updated Final Safety Analysis Report (UFSAR) (Reference 1) identifies the non-loss-of-coolant accident (non-LOCA) transient events that have been analyzed as part of the current Cook Unit 1 licensing basis. In support of the Return to Reactor Coolant System (RCS) Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) Program, the non-LOCA licensing basis events, as well as the Anticipated Transient without Scram (ATWS) event, were either evaluated or re-analyzed to demonstrate the acceptability of bounding NOP/NOT operating conditions with respect to the non-LOCA licensing basis analyses. The conditions considered correspond to those in Table 2.1-1, which support up to 30 percent steam generator tube plugging (SGTP) and a thermal design flow (TDF) that is consistent with the current licensing basis non-LOCA analyses. It is noteworthy that the NOP/NOT operating conditions considered for the non-LOCA analyses are more conservative and/or bound a larger range than those considered in some other analysis areas (e.g., 30 percent SGTP is considered in the non-LOCA analyses, whereas 10 percent SGTP is considered for other analysis areas).

The current non-LOCA licensing basis analyses for Cook Unit 1 were performed to bound operation for both the reduced and normal operating pressures and the entire allowable vessel average temperature (T_{avg}) window; however, since the time when Cook Unit 1 moved to a lower operating temperature and pressure, evaluations for various events have been performed taking credit for the lower temperature and/or pressure. In particular were the overtemperature ΔT (OTΔT) and overpower ΔT (OPΔT) setpoints, which utilized T' and T'' values that were restricted below the full power T_{avg} primarily to provide overpower protection while maintaining the same ΔT setpoints (Reference 6).

Additionally, as discussed in Section 6.2, revised fuel temperature data was calculated as part of the Return to RCS NOP/NOT Program based on current fuel performance and design models (PAD 4.0). The minimum fuel temperatures calculated for the program were more limiting than those modeled in the analyses of record (AOR); therefore, events performed using the LOFTRAN code that modeled minimum fuel temperatures (i.e., maximum fuel heat transfer characteristics, primarily departure from nucleate boiling (DNB) events) required evaluation.

Finally, an increased low pressurizer pressure reactor trip nominal setpoint was defined for the Return to RCS NOP/NOT Program and was evaluated to confirm that there was no adverse impact on the non-LOCA safety analyses.

The primary non-LOCA scope for the Return to RCS NOP/NOT Program was to perform an explicit steam line break – core response from hot full power steam line break (HFP SLB) analysis to demonstrate that adequate overpower protection exists with the restriction on T'' removed, and to evaluate that the results of interest for all other events continue to remain bounding with respect to the revised operating conditions and fuel temperature data. The HFP SLB analysis is discussed in subsection 5.2.2, and the evaluations for the remaining non-LOCA events are discussed in subsection 5.2.3.

The non-LOCA analyses and evaluations for the Return to RCS NOP/NOT Program, with the exception of the HFP SLB analysis, were performed using codes and methods consistent with the analyses that support the current Cook Unit 1 licensing basis. As discussed in subection 5.2.2, a HFP case for the SLB event had not been previously analyzed. The application of the Westinghouse methodology for the HFP SLB is discussed in subection 5.2.2.

5.2.2 Analyzed Events (Hot Full Power Steam Line Break)

5.2.2.1 Background

The only non-LOCA event that required a full analysis was the SLB event initiated from HFP. The analysis of the SLB event traditionally assumed Mode 2 conditions. The greatest cooldown, and therefore the greatest reactivity excursion, would occur from Mode 2 conditions, where the decay heat level is low and the steam generator (SG) shell side inventory and pressure are high. However, full-power conditions may challenge the ability of the OPΔT protection function to protect against the nuclear fuel overpower limit; therefore, the analysis is also performed at full power to demonstrate that a reactor trip is demanded by the reactor trip system (RTS) and executed in time to provide adequate protection to preclude fuel damage in order to confirm safe shutdown during Mode 1 operation. Once the reactor is tripped, the potential for fuel and cladding damage and fuel centerline melting during the remainder of the transient would be addressed by the hot zero power (HZP) analysis evaluated in subsection 5.2.3.

5.2.2.2 Input Parameters and Assumptions

The following summarizes the major input parameters and/or assumptions used in the analysis of the HFP SLB event:

1. This accident is analyzed with the Revised Thermal Design Procedure (RTDP) (Reference 5). Initial reactor power, RCS pressure, and RCS temperature are assumed to be at their nominal values, consistent with steady-state full power operation.

2. Minimum measured flow (MMF) is modeled.

3. The analysis assumed maximum moderator reactivity feedback and minimum Doppler power feedback to maximize the power increase following the break.

4. In computing the steam flow during a steam line rupture, the Moody Curve for f(L/D) = 0 is used.

5. Maximum T_{avg} value was assumed.

6. 0-percent SGTP is assumed to maximize the primary-to-secondary heat transfer rate.

7. The analysis assumes up to a complete severance of a main steam pipe with the plant initially at full-power conditions. Since the Babcock & Wilcox International (BWI) Model 51 SGs are equipped with integral flow restrictors with a 1.4 ft^2 throat area, any steam line rupture with a break area greater than this size, regardless of the location, would have the same effect on the reactor as a 1.4 ft^2 break.

8. Pressurizer heaters are not credited. This assumption conservatively yields a higher rate of pressure decrease.

9. The pressurizer power-operated relief valves (PORVs) are modeled to reduce RCS pressure, resulting in a conservative calculation of the margin to the departure from nucleate boiling ratio (DNBR) limit.

10. The protection system feature that mitigates the effects of a SLB initiated from HFP is a reactor trip, if necessary (specifically, OPΔT and low steam pressure – safety injection (SI)).

11. For the low steam pressure – SI reactor trip function, the modeled setpoint included quantified environmental uncertainties.

5.2.2.3 Acceptance Criteria

Depending on the break size, the HFP SLB event can be considered either an American Nuclear Society (ANS) Condition III or IV event. However, some minor steam line breaks are indistinguishable from credible breaks such as Inadvertent Opening of a Main Steam Safety Valve or Inadvertent Actuation of Steam Dump, which are Condition II events, and in order to bound these cases, Condition II criteria must be satisfied. Therefore, Condition II criteria are conservatively applied for all break sizes analyzed for ease of interpretation.

The specific acceptance criteria applied in the analysis are as follows:

- DNBR should remain above the 95/95 DNBR limit at all times during the transient. In addition, the peak linear heat generation rate (expressed in kW/ft) should not exceed a value that would cause fuel centerline melt.

- Primary and secondary pressures must remain below 110 percent of the respective design pressures at all times during the transient. However, since this event results in a decrease in both the primary and secondary side pressures, these criteria are not challenged by the HFP SLB event.

5.2.2.4 Description of Analyses and Evaluations

A detailed analysis using the LOFTRAN computer code (Reference 2) is performed to determine the plant transient conditions following a main steam line rupture at HFP conditions. The code computes pertinent variables, including the core power, RCS temperature, and pressure. A spectrum of break sizes is analyzed, and a limiting case is determined based on the maximum peak core heat flux calculated. Statepoints for the limiting case consisting of nuclear power, RCS loop inlet temperatures, pressures, and core flow, along with additional inputs from the nuclear core model analyzed using the ANC code (Reference 3), are used as input to the detailed thermal and hydraulic digital computer code VIPRE (Reference 4) to determine if the DNBR design basis is satisfied for the limiting time in the transient. The DNBR calculations were performed using the WRB-1 DNB correlation and RTDP (Reference 5). Additionally, the nuclear core model analyzed using ANC code determines if the maximum peak linear heat generation rate limit (expressed in kW/ft) is violated. Cladding stress and strain are evaluated on a cycle-by-cycle basis.

5.2.2.5 Results

The most limiting HFP SLB is the case for the largest break size for which a reactor trip on OPΔT is predicted. This is because for the cases assuming smaller break sizes, no reactor trip is predicted and the power reaches a non-limiting equilibrium at the higher steam load, and for the cases assuming larger break sizes, the reactor trips relatively quickly on a SI signal due to low steam line pressure, making the larger break sizes less limiting due to the reactor trip occurring before a significant power excursion can occur. Based on this, the limiting case for NOP/NOT conditions is a 0.89 ft^2 break. As discussed in Sections 6.1 and 6.3 of this report, statepoints for this limiting case were analyzed using the ANC and VIPRE codes, and it was determined that the DNBR design basis was satisfied and the maximum peak linear heat generation rate remained below the level that would result in fuel centerline melt.

The sequence of events for the limiting case is shown in Table 5.2.2-1. Figures 5.2.2-1 through 5.2.2-4 show the transient results.

5.2.2.6 Conclusions

It is concluded that the analysis of the HFP SLB event has adequately accounted for operation of the plant at NOP/NOT conditions and was performed using acceptable analytical models. Furthermore, it is concluded that the analysis has demonstrated that the reactor protection and safety systems ensure that the specified acceptable fuel design limits will not be exceeded as a result of a HFP SLB event. Therefore, operation at NOP/NOT conditions is acceptable with respect to the HFP SLB event.

5.2.3 Evaluated Events

Each of the UFSAR transients, as well as ATWS, listed in Table 5.2.3-1, was evaluated in support of the Cook Unit 1 Return to RCS NOP/NOT Program.

Four different methods were used to evaluate the events: events for which the current analysis bounded operation at NOP/NOT conditions, events which required some analysis to demonstrate that the current analysis bounded operation at NOP/NOT conditions, events which are addressed by the plant Technical Specifications (TS) and thus do not require evaluation, and events that were fully reanalyzed (HFP SLB only, refer to subsection 5.2.2). The notes in Table 5.2.3-1 indicate the method(s) used to address each event.

Note that the non-LOCA analyses support a variety of operating conditions, with some analyses conservatively supporting maximum conditions that are greater than the maximum allowable current operating conditions (e.g., some analyses support a maximum T_{avg} of 576.3°F or a maximum NSSS power level of 3409 MWt).

5.2.3.1 Uncontrolled Rod Withdrawal from Subcritical – UFSAR Chapter 14.1.1

The licensing basis analysis of the Uncontrolled Rod Withdrawal from Subcritical (RWFS) event presented in UFSAR Chapter 14.1.1 was performed assuming an initial pressurizer pressure consistent with operation at the reduced operating pressure and the no-load temperature, the latter of which is unchanged for the Return to RCS NOP/NOT Program. Since the primary acceptance criterion for the

RWFS event is to demonstrate that fuel damage is precluded by satisfying the DNB design basis, a lower initial primary pressure is conservative; therefore, the analysis would realize additional margin through modeling the increased primary pressure of NOP/NOT conditions. As a result, the current licensing basis analysis for the RWFS event bounds operation at NOP/NOT conditions and the conclusions presented in UFSAR Chapter 14.1.1 remain valid.

5.2.3.2 Uncontrolled Rod Withdrawal at Power – UFSAR Chapter 14.1.2

The primary acceptance criteria for the Uncontrolled Rod Withdrawal at Power (RWAP) event are demonstrating that the DNB design basis is satisfied and that the primary pressure boundary is protected (peak RCS pressure remains below 110 percent of the design pressure).

The licensing basis analysis for the RWAP event presented in UFSAR Chapter 14.1.2 was performed assuming initial conditions that are consistent with or bounding of NOP/NOT conditions. However, the DNB cases, which model minimum fuel temperatures, required evaluation. To address the revised minimum fuel temperatures for the Return to RCS NOP/NOT Program, the limiting RWAP DNB cases were evaluated using the LOFTRAN code modeling the revised fuel heat transfer characteristics and an initial pressurizer pressure of 2250 psia (versus the previously modeled 2100 psia). The evaluation demonstrated that the limiting statepoint for the NOP/NOT evaluation was less limiting than that of the current licensing basis analysis, primarily due to the increased operating pressure. Additionally, it was confirmed as part of the Return to RCS NOP/NOT Program that the OTΔT setpoints modeled in the current analysis remain valid at NOP/NOT conditions.

Therefore, the current licensing basis analysis for the RWAP event bounds operation at NOP/NOT conditions and the conclusions presented in UFSAR Chapter 14.1.2 remain valid.

5.2.3.3 Rod Cluster Control Assembly Misalignment / Rod Cluster Control Assembly Drop – UFSAR Chapters 14.1.3/14.1.4

The licensing basis analysis of the Rod Cluster Control Assembly (RCCA) Misalignment and RCCA Drop events presented in UFSAR Chapters 14.1.3 and 14.1.4 were performed assuming a range of initial conditions that are consistent with or bounding of NOP/NOT conditions. Since the primary acceptance criterion for the RCCA Misalignment and RCCA Drop events is to demonstrate that fuel damage is precluded by satisfying the DNB design basis, a lower initial primary pressure is conservative; therefore, the analysis would realize additional margin through modeling the increased primary operating pressure of NOP/NOT conditions. Additionally, the applicability of the statepoints of the analysis has been confirmed for a maximum T_{avg} that bounds that of the Return to RCS NOP/NOT Program, thus additional margin would be realized through modeling the NOP/NOT maximum T_{avg}. Therefore, the current licensing basis analysis for the RCCA Misalignment and RCCA Drop events bounds operation at NOP/NOT conditions and the conclusions presented in UFSAR Chapters 14.1.3 and 14.1.4 remain valid.

5.2.3.4 Uncontrolled Boron Dilution – UFSAR Chapter 14.1.5

The licensing basis analyses of the Uncontrolled Boron Dilution event presented in UFSAR Chapter 14.1.5 were performed to demonstrate that a loss of shutdown margin is precluded for all modes of operation. The analysis was performed using reactor coolant density values based on the lower operating pressure and a maximum T_{avg} that bounds that of the Return to RCS NOP/NOT Program, conservatively minimizing the initial RCS inventory to be diluted. Thus, while the effect would be minor, increasing the primary operating pressure and reducing the maximum T_{avg} to the NOP/NOT values would increase the initial RCS inventory, providing additional time for operators to prevent a loss of shutdown margin. Therefore, the current licensing basis analyses for the Uncontrolled Boron Dilution event bounds operation at NOP/NOT conditions and the conclusions presented in UFSAR Chapters 14.1.5 remain valid.

5.2.3.5 Loss of Flow – UFSAR Chapter 14.1.6

The licensing basis analysis of the Loss of Flow (LOF) event presented in UFSAR Chapter 14.1.6 was performed to primarily demonstrate that the DNB design basis is satisfied. The analysis was performed assuming initial conditions that are consistent with or bounding of NOP/NOT conditions. However, the analysis modeled minimum fuel temperatures and thus required evaluation. To address the revised minimum fuel temperatures for NOP/NOT, the limiting LOF case (corresponding to the complete LOF event tripping on an undervoltage signal) was evaluated using the LOFTRAN code modeling the revised fuel heat transfer characteristics and an initial pressurizer pressure of 2250 psia (versus the previously modeled 2100 psia).

The evaluation demonstrated that the limiting statepoint for the NOP/NOT evaluation was less limiting than that of the current licensing basis analysis, primarily due to the increased operating pressure. Therefore, the current licensing basis analysis for the LOF event bounds operation at NOP/NOT conditions and the conclusions presented in UFSAR Chapter 14.1.6 remain valid.

5.2.3.6 Locked Rotor – UFSAR Chapter 14.1.6

The licensing basis analyses of the Locked Rotor event presented in UFSAR Chapter 14.1.6 were performed to determine what percentage (if any) of rods are expected to be in DNB during the transient, that the peak clad temperature remains below the limit value, and that the RCS pressure boundary is protected (pressure remains below 110 percent of the design value). The analysis was performed assuming initial conditions that are consistent with or bounding of NOP/NOT conditions; however, the Rods-in-DNB cases, which model minimum fuel temperatures, required evaluation. To address the revised minimum fuel temperatures for NOP/NOT, the Locked Rotor Rods-in-DNB case was evaluated using the LOFTRAN code modeling the revised fuel heat transfer characteristics, an initial pressurizer pressure of 2250 psia (versus the previously modeled 2100 psia), and a T_{avg} consistent with the maximum for the NOP/NOT operating conditions (575.4°F).

The evaluation demonstrated that the limiting statepoint for the NOP/NOT evaluation was less limiting than that of the current licensing basis analysis, primarily due to the increased operating pressure. Therefore, the current licensing basis analysis for the Locked Rotor event bounds operation at NOP/NOT conditions and the conclusions presented in UFSAR Chapter 14.1.6 remain valid.

5.2.3.7 Startup of an Inactive Loop – UFSAR Chapter 14.1.7

Per the discussion in UFSAR Chapter 14.1.7, the Startup of an Inactive Loop (SUIL) event is precluded due to the plant TSs prohibiting operation with an idle reactor loop. As a result, no analysis is performed for this event and the basis for addressing the SUIL event presented in UFSAR Chapter 14.1.7 remains valid for operation at NOP/NOT conditions.

5.2.3.8 Loss of Load – UFSAR Chapter 14.1.8

The licensing basis analysis of the Loss of Load (LOL) event presented in UFSAR Chapter 14.1.8 was performed assuming initial conditions that are consistent with or bounding of NOP/NOT conditions. The primary acceptance criteria for the event are to demonstrate that the DNB design basis is satisfied and that primary pressure remains below 110 percent of the design value.

Separate sets of cases were performed in the licensing basis analysis to demonstrate that each criterion was satisfied. Each set of cases modeled conditions that were consistent with or bounding of NOP/NOT conditions, with the primary pressure case modeling a T_{avg} that bounds NOP/NOT conditions and both the primary pressure and DNB cases modeling the lower primary pressure. Both cases would realize additional margin through modeling the increased primary operating pressure, and the primary pressure case would realize further margin through the reduction in T_{avg}. Therefore, the current licensing basis analysis for the LOL event bounds operation at NOP/NOT conditions and the conclusions presented in UFSAR Chapter 14.1.8 remain valid.

5.2.3.9 Loss of Normal Feedwater/Loss of AC Power – UFSAR Chapter 14.1.9

The licensing basis analyses of the Loss of Normal Feedwater/Loss of AC Power (LONF/LOAC) events presented in UFSAR Chapters 14.1.9 and 14.1.12 were performed to demonstrate that the transient does not propagate into a more serious event by showing that the pressurizer does not become water solid. The analysis modeled initial conditions that are consistent with or bounding of NOP/NOT conditions, particularly the modeled initial NSSS power level of 3409 MWt plus two percent uncertainty. Therefore, the current licensing basis analysis for the LONF/LOAC events bounds operation at NOP/NOT conditions and the conclusions presented in UFSAR Chapters 14.1.9 and 14.1.12 remain valid.

5.2.3.10 Feedwater Malfunction – UFSAR Chapter 14.1.10

The licensing basis analysis of the Feedwater Malfunction (FWM) event presented in UFSAR Chapter 14.1.10 was performed to primarily demonstrate that the DNB design basis is satisfied. This event includes cases initiated from both zero-power (for the reduction in feedwater temperature case) and full-power (for the increase in feedwater flow) conditions.

The licensing basis analysis for all FWM cases presented in UFSAR Chapter 14.1.10 was performed assuming initial conditions that are consistent with or bounding of NOP/NOT conditions. However, the analysis modeled minimum fuel temperatures and thus required evaluation. To address the revised minimum fuel temperatures for NOP/NOT, the limiting HFP and HZP cases were evaluated using the LOFTRAN code modeling the revised fuel heat transfer characteristics and an initial pressurizer pressure of 2250 psia (versus the previously modeled 2100 psia). The results demonstrated that NOP/NOT evaluations were less limiting than that of the current licensing basis analysis, primarily due to the increased operating pressure. The HFP case yielded less limiting statepoints, while the HZP case resulted in a lower maximum reactivity insertion rate compared to the current licensing basis analysis.

Therefore, the current licensing basis analysis for the FWM event bounds operation at NOP/NOT conditions and the conclusions presented in UFSAR Chapter 14.1.10 remain valid.

5.2.3.11 Excessive Load Increase – UFSAR Chapter 14.1.11

The licensing basis analysis of the Excessive Load Increase (ELI) events presented in UFSAR Chapter 14.1.11 is performed to demonstrate that the DNB design basis is satisfied. The licensing basis analysis modeled conditions consistent with or bounding of NOP/NOT conditions and the analysis would realize additional margin if the increased primary operating pressure were modeled. Therefore, the current licensing basis analysis for the ELI event bounds operation at NOP/NOT conditions and the conclusions presented in UFSAR Chapter 14.1.11 remain valid.

5.2.3.12 Rupture of a Steam Pipe – UFSAR Chapter 14.2.5

The licensing basis analysis of the SLB event presented in UFSAR Chapter 14.2.5 was performed to primarily demonstrate that the DNB design basis is satisfied and that the peak linear heat generation rate does not exceed a value that would result in fuel centerline melt. The current licensing basis analysis was performed from HZP conditions; refer to subsection 5.2.2 for the analysis of the SLB event from HFP conditions performed at NOP/NOT conditions.

The licensing basis analysis for the SLB event from HZP presented in UFSAR Chapter 14.2.5 was performed assuming initial conditions that are consistent with or bounding of NOP/NOT conditions. However, the analysis modeled minimum fuel temperatures and thus required evaluation. To address the revised minimum fuel temperatures for NOP/NOT, the limiting HZP case (double-ended rupture located downstream of the flow restrictor with offsite power available) was evaluated using the LOFTRAN code modeling the revised fuel heat transfer characteristics and an initial pressurizer pressure of 2250 psia (versus the previously modeled 2100 psia). The results demonstrated that the statepoint values for the NOP/NOT evaluation were less limiting than those of the current licensing basis analysis, primarily due to the increased operating pressure.

Therefore, the current licensing basis analysis for the SLB event initiated from HZP bounds operation at NOP/NOT conditions and the conclusions for the HZP case presented in UFSAR Chapter 14.2.5 remain valid. Refer to subsection 5.2.2 for discussion of the HFP case.

5.2.3.13 Rod Cluster Control Assembly Ejection – UFSAR Chapter 14.2.6

The licensing basis analysis of the RCCA Ejection event presented in UFSAR Chapter 14.2.6 is performed to demonstrate that severe fuel damage is precluded. This criterion is satisfied by showing that the peak fuel centerline temperature, peak fuel average temperature, maximum fuel energy peak clad average temperature, percent fuel melted and maximum zirconium-water reaction are maintained below acceptable limits. The licensing basis analysis modeled conditions consistent with or bounding of NOP/NOT conditions, and, based on the results of sensitivities that support the current licensing basis analysis, the analysis would realize additional margin if the increased primary operating pressure was modeled. Therefore, the current licensing basis analysis for the RCCA Ejection event bounds operation at NOP/NOT conditions and the conclusions presented in UFSAR Chapter 14.2.6 remain valid.

5.2.3.14 Anticipated Transient Without Scram

The ATWS event analysis is a beyond licensing basis analysis for Cook Unit 1. Consistent with the currently employed approach, ATWS concerns for Cook Unit 1 are addressed through demonstrating that the calculated unfavorable exposure time (UET), which is the period of the fuel cycle time (calculated on a cycle-by-cycle basis) when the moderator temperature coefficient is insufficiently negative to maintain RCS pressure below the American Society of Mechanical Engineers (ASME) Service Level C limit, is restricted to less than 5 percent of the operating cycle. To determine UET, the reactivity feedback conditions of the core and plant conditions on a cycle-by-cycle basis must be compared to the ATWS analysis conditions that lead to a peak RCS pressure consistent with the ASME Service Level C limit. The heatup and shutdown characteristics, referred to as critical power trajectories (CPTs), for a specific plant configuration (primarily total reactivity feedback, primary-side pressure relief capacity, and auxiliary feedwater capacity) are calculated. The analysis that generated the CPTs modeled conditions which are consistent with or bounding of NOP/NOT conditions, and would realize margin through less limiting heatup/shutdown characteristics through modeling the lower NOP/NOT maximum T_{avg}. Therefore, the current analysis continues to be valid for addressing ATWS concerns for operation at NOP/NOT conditions.

5.2.3.15 Revised Low Pressurizer Pressure Reactor Trip Nominal Setpoint

The low pressurizer pressure reactor trip is considered in the analyses of the RCCA Misalignment/RCCA Drop and ELI events. An increase in the nominal setpoint, as identified for the Return to RCS NOP/NOT Program, does not necessitate a change to the non-LOCA safety analysis setpoint. Any increase in the trip setpoint would be a benefit to the non-LOCA safety analyses as it would result in an earlier reactor trip if required. The low pressurizer pressure reactor trip is not actuated in any of the licensing basis non-LOCA safety analyses for Cook Unit 1; thus, there is no impact on the non-LOCA licensing basis analyses.

5.2.3.16 Evaluated Events Conclusions

It is concluded that the analyses for all events presented in Table 5.2.3-1 can accommodate operation at NOP/NOT conditions, including addressing the revised minimum fuel temperature data.

5.2.3.17 References

1. "D. C. Cook Nuclear Plant Updated Final Safety Analysis Report," Revision 24.0, March 2012.

2. Westinghouse Report WCAP-7907-P-A, "LOFTRAN Code Description," April 1984.

3. Westinghouse Report WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1986.

4. Westinghouse Report WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

5. Westinghouse Report WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

6. Letter from J. E. Pollock (AEP) to USNRC, "Donald C. Cook Nuclear Plant Unit 1 Docket No. 50-315 License Amendment Request for Appendix K Measurement Uncertainty Recapture – Power Uprate Request," dated June 28, 2002. (Available in NRC ADAMS under Accession Number ML030990129)

Table 5.2.2-1 Time Sequence of Events – HFP SLB

Case	Event	Time (sec)
Largest break size that trips on OPΔT – 0.89 ft^2 break	Steam line rupture	0.0
	OPΔT reactor trip setpoint reached (4 loops)	22.6
	Rod motion initiated	24.6
	Peak core heat flux occurs	25.2
	Minimum DNBR occurs	25.4

Table 5.2.3-1 Non-LOCA Licensing Basis Events for Cook Unit 1

Event	Report Subsection	UFSAR Chapter	Notes
Uncontrolled Rod Cluster Control Assembly Withdrawal from a Subcritical Condition	5.2.3.1	14.1.1	1
Uncontrolled RCCA Withdrawal at Power	5.2.3.2	14.1.2	1 (overpressure) 2 (DNB)
RCCA Misalignment RCCA Drop	5.2.3.3	14.1.3 14.1.4	1
Chemical and Volume Control System Malfunction (Uncontrolled Boron Dilution)	5.2.3.4	14.1.5	1
Loss of Reactor Coolant Flow	5.2.3.5	14.1.6	2
Locked Rotor	5.2.3.6	14.1.6	1 (overpressure) 2 (rods-in-DNB)
Startup of an Inactive Reactor Coolant Loop	5.2.3.7	14.1.7	3
Loss of External Electrical Load	5.2.3.8	14.1.8	1
Loss of Normal Feedwater Flow Loss of All AC Power to the Plant Auxiliaries	5.2.3.9	14.1.9 14.1.12	1
Excessive Heat Removal Due to Feedwater System Malfunctions	5.2.3.10	14.1.10	2
Excessive Load Increase Incident	5.2.3.11	14.1.11	1
Rupture of a Steam Pipe	5.2.3.12 (HZP) 5.2.2 (HFP)	14.2.5	1 (HZP) 4 (HFP)
Rupture of Control Rod Drive Mechanism Housing (RCCA Ejection)	5.2.3.13	14.2.6	1
Anticipated Transient Without Scram	5.2.3.14	n/a	1

Notes:
1. Current analysis bounds NOP/NOT conditions
2. Evaluation performed to demonstrate that current analysis is bounding of NOP/NOT conditions
3. Transient is addressed through the plant TSs; no analysis required
4. Complete analysis performed for the HFP case of the SLB event; refer to subsection 5.2.2

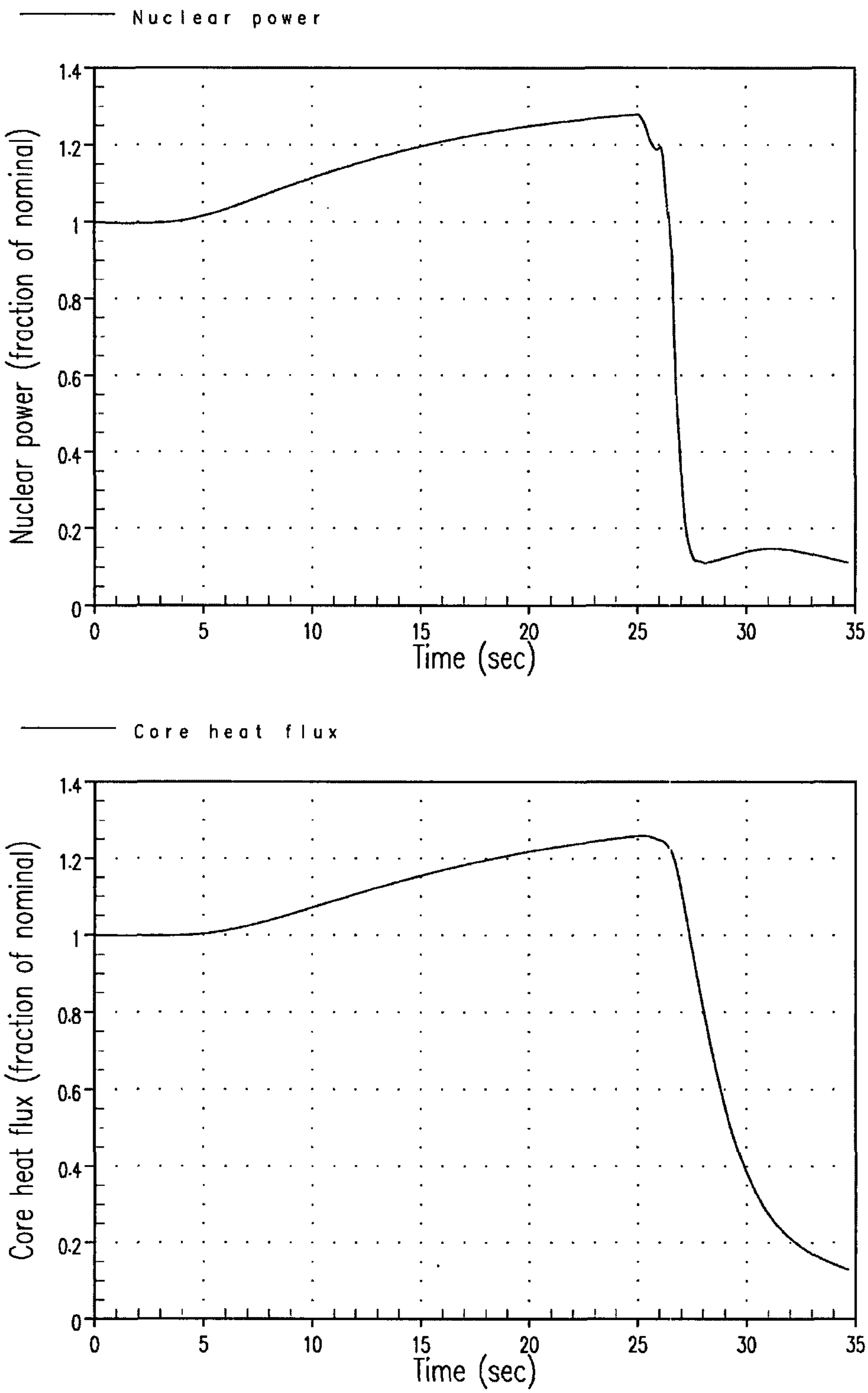
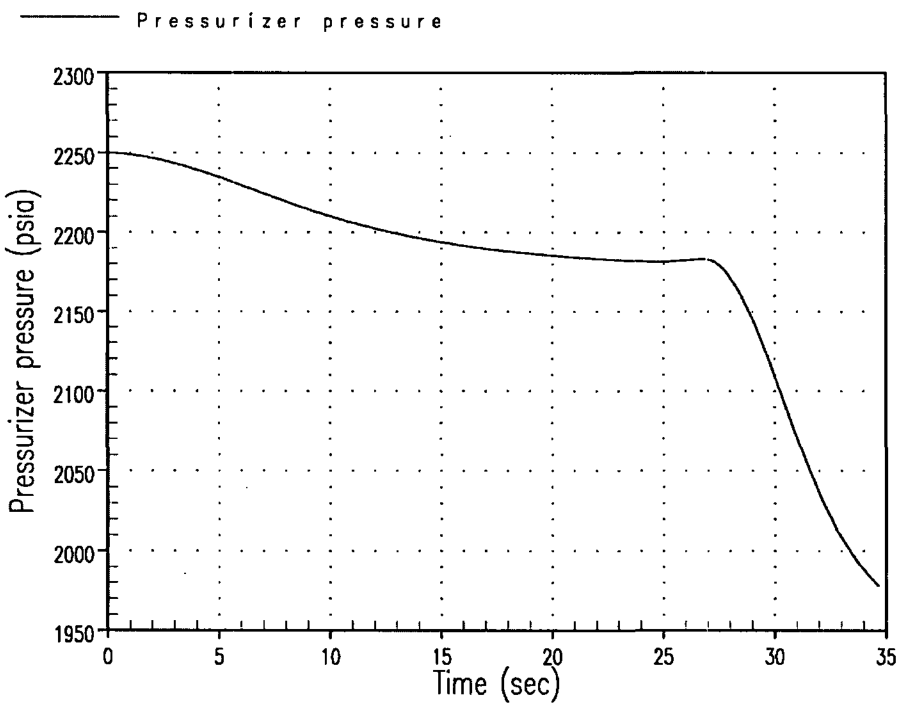
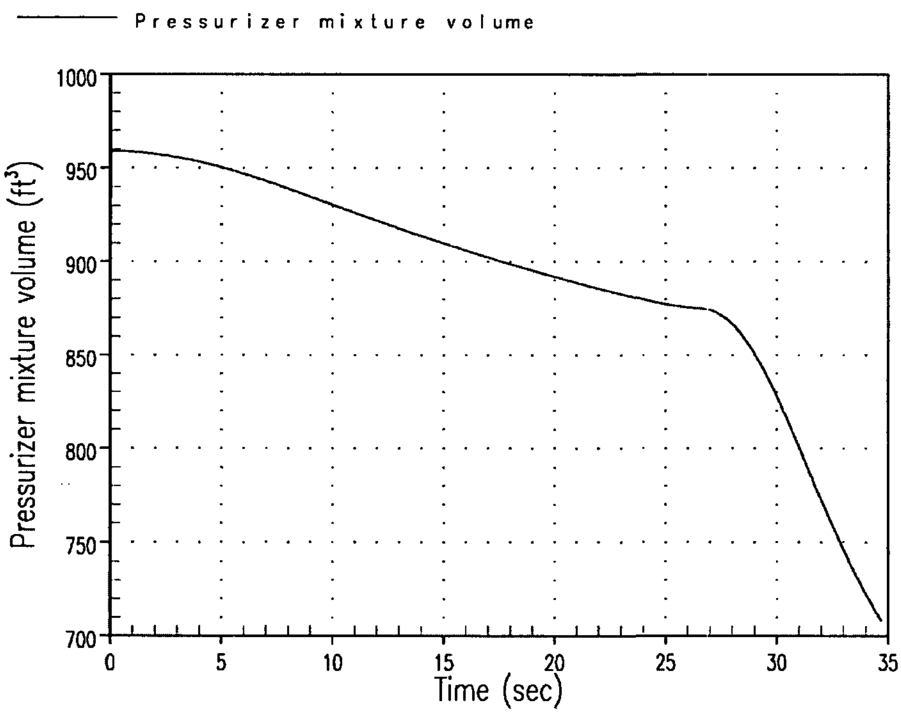


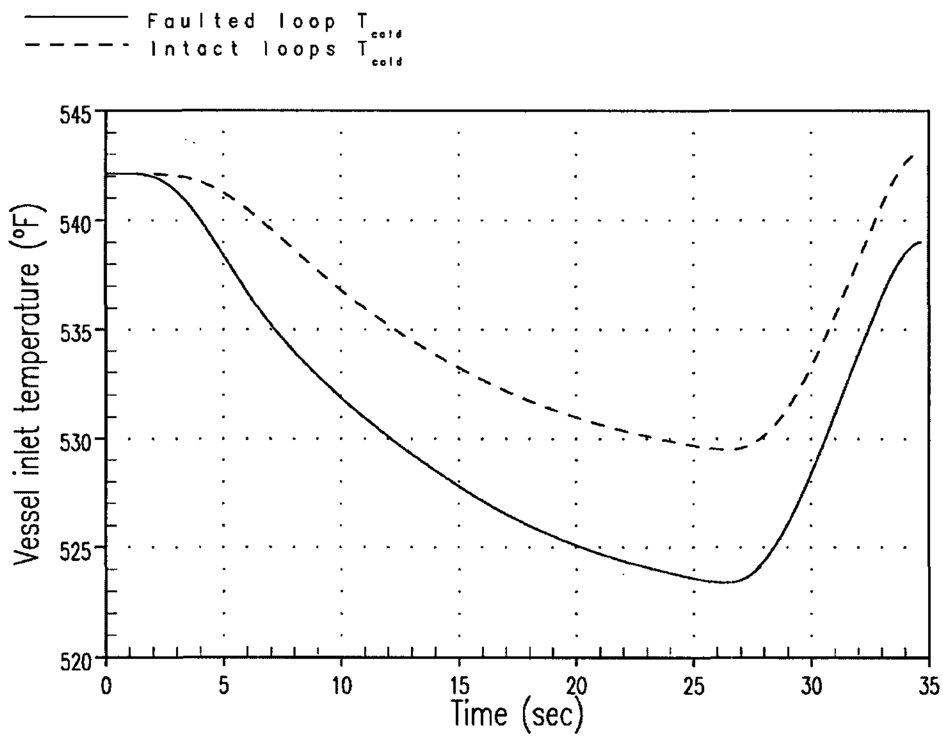
Figure 5.2.2-1 Steam System Piping Failure at Full Power – 0.89 ft^2 Break Nuclear Power and Core Heat Flux versus Time





Note that the pressurizer mixture volume includes the pressurizer surgeline volume.

Figure 5.2.2-2 **Steam System Piping Failure at Full Power – 0.89 ft^2 Break Pressurizer Pressure and Pressurizer Mixture Volume versus Time**



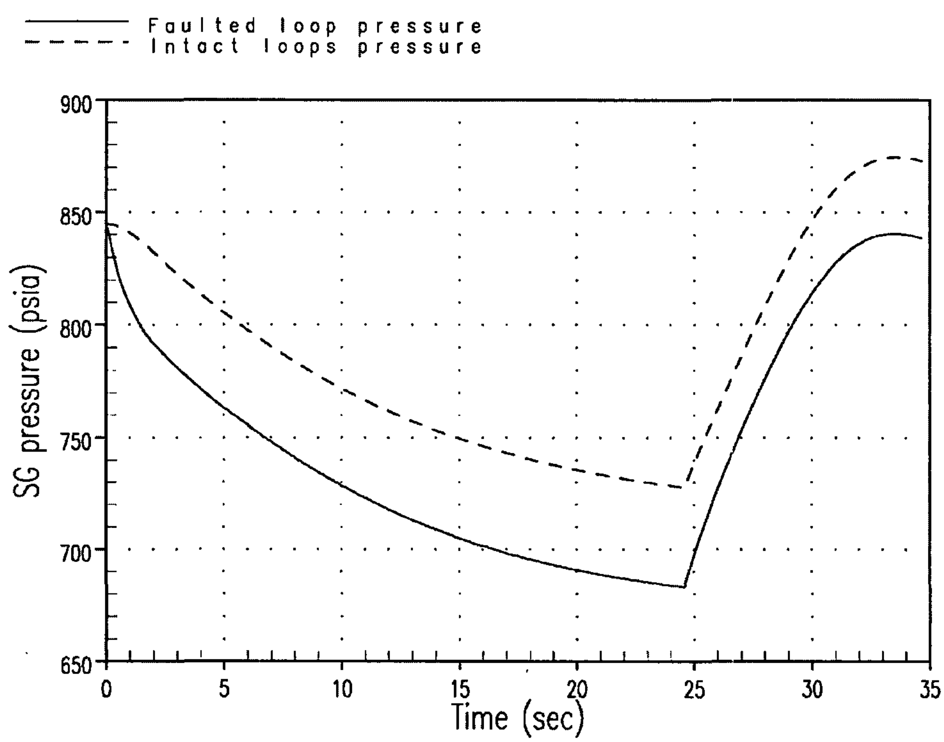


Figure 5.2.2-3 Steam System Piping Failure at Full Power – 0.89 ft^2 Break Vessel Inlet Temperature and SG Pressure versus Time

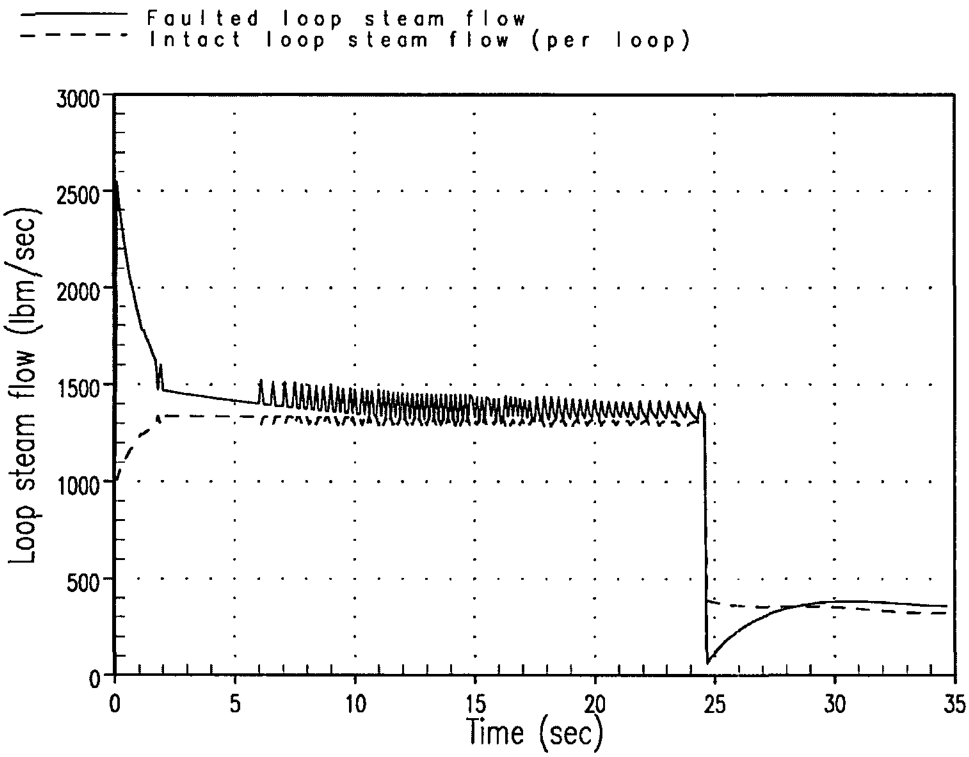


Figure 5.2.2-4 **Steam System Piping Failure at Full Power – 0.89 ft^2 Break SG Outlet Steam Flow Rate versus Time**

5.3 STEAM GENERATOR TUBE RUPTURE (SGTR)

5.3.1 Introduction

The Cook Unit 1 analysis of the steam generator tube rupture (SGTR) is described in Updated Final Safety Analysis Report (UFSAR) Chapter 14.2.4, Steam Generator Tube Rupture. The SGTR accident is a double-ended break of one steam generator (SG) tube that results in the transfer of primary coolant to the secondary side of the affected SG. The SGTR analysis conservatively models the primary-to-secondary break flow and steam releases from the ruptured SG to the environment for 30 minutes using a simplified "hand calculation." This analysis provides input to the calculation of the offsite and control room dose consequences of the SGTR. This simplified analysis does not specifically model the operator responses to the SGTR and does not address the potential for SG overfill. These aspects of the SGTR analysis, which consider the additional time needed to meet current Emergency Operating Procedure (EOP) requirements, were addressed in a supplemental analysis that was the subject of a License Amendment Request (Reference 1) in 2000, a response to an USNRC Request for Additional Information (Reference 2), and approved Units 1 and 2 License Amendments in 2001 (Reference 3). The supplemental analyses confirmed the accuracy of the implicit assumptions in the simplified hand calculation that the SGs would not overfill and, as a result, the radiological consequences calculations do not need to consider liquid water release from the ruptured SG.

The impact of the Return to Reactor Coolant System (RCS) Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) Program on each of these three SGTR analyses was examined. subsection 5.3.2 describes the review of NOP/NOT on the licensing basis input to the dose analysis. subsection 5.3.3 addresses the impact of the Return to RCS NOP/NOT Program on the conclusion that the licensing basis input to dose analysis (discussed in subsection 5.3.2) provides conservative input to the dose analysis even though it only models 30 minutes of break flow. Subsection 5.3.4 addresses NOP/NOT impacts on margin to overfill (MTO) and the related possibility that liquid water could be released from the ruptured SG.

5.3.1.1 References

1. C1000-11, "Donald C. Cook Nuclear Plant Units 1 and 2 License Amendment for Changes in Steam Generator Tube Rupture Analysis Methodology," dated October 24, 2000. (Available in NRC ADAMS under Accession Number ML003762982)

2. C0601-21, "Donald C. Cook Nuclear Plant Units 1 and 2 Response to Nuclear Regulatory Commission Request for Additional Information Regarding License Amendment for Changes in Steam Generator Tube Rupture Analysis Methodology (TAC Nos. MB0739 and MB0740)," dated June 29, 2001. (Available in NRC ADAMS under Accession Number ML011860097)

3. Letter from John F. Stang (U.S. NRC) to Robert P. Powers (AEP), "Donald C. Cook Nuclear Plant, Units 1 and 2 – Issuance of Amendments (TAC Nos. MB0739 and MB0740)," dated October 21, 2001. (Available in NRC ADAMS under Accession Number ML012690136)

5.3.2 SGTR – Licensing Basis Input to Dose

5.3.2.1 Introduction and Background

The major hazard associated with an SGTR event is the radiological consequences resulting from the transfer of radioactive primary coolant to the secondary side of the ruptured SG and subsequent release of radioactivity to the atmosphere. The Cook Unit 1 licensing basis SGTR input to dose analysis determines the mass releases for use in calculating the radiological consequences without ruptured SG overfill. The licensing basis SGTR input to dose analysis described in UFSAR Chapter 14.2.4 is a simplified mass and energy balance that calculates the primary to secondary break flow for 30 minutes and steam releases to the atmosphere for use in the SGTR radiological consequence analysis.

5.3.2.2 Input Parameters and Assumptions

The licensing basis SGTR input to dose analysis which supports the UFSAR for Cook Unit 1 will remain bounding for NOP/NOT as long as the following input assumptions used in the UFSAR analysis are met:

1. T_{avg} is $\geq$ 553.7°F and $\leq$ 575.4°F

2. RCS pressure is $\geq$ 2100 psia and $\leq$ 2250 psia

5.3.2.3 Acceptance Criteria

The current licensing basis analyses were performed to calculate the mass transfer data for input to the SGTR radiological consequences analysis. As such, the NOP/NOT conditions must be bounded by the current licensing basis SGTR input to dose analysis in order for the SGTR doses presented in UFSAR Chapter 14.2.4.5 to remain valid.

5.3.2.4 Description of Analyses and Evaluations

The input parameters which support operation of Cook Unit 1 at NOP/NOT conditions are presented in the Introduction (Section 1). The Section 1 conditions were compared with the input parameters which support the licensing basis SGTR input to dose analysis of record to determine if the analysis is still valid at NOP/NOT conditions.

5.3.2.5 Results

The current licensing basis SGTR input to dose analysis is analyzed at a T_{avg} between 553.7°F and 575.4°F and a pressure of 2100 or 2250 psia. The input parameters in Section 1 show a T_{avg} of 571.0°F and pressure of 2250 psia for NOP/NOT. Therefore, the temperature and pressure conditions of NOP/NOT are bounded and no new licensing basis SGTR input to dose analysis is required.

5.3.2.6 Conclusions

The current licensing basis SGTR input to dose analysis remains applicable at NOP/NOT conditions. As such, the SGTR doses presented in UFSAR Chapter 14.2.4.5 remain valid.

5.3.3 SGTR – Supplemental Input to Dose

5.3.3.1 Introduction and Background

The supplemental SGTR input to dose analysis calculates mass releases with a more realistic response to an SGTR event than that considered in the licensing basis SGTR input to dose analysis. This includes modeling operator actions which lead to break flow termination more than 30 minutes after accident initiation. The supplemental SGTR input to dose analysis is used to determine if the licensing basis SGTR input to dose analysis is conservative, even though it assumes that break flow persists for only 30 minutes from the initiation of the event.

The supplemental SGTR input to dose analysis is described in Reference 1.

5.3.3.2 Input Parameters and Assumptions

The supplemental SGTR input to dose analysis was evaluated using the following NOP/NOT conditions:

1. T_{avg} of 571.0°F
2. RCS pressure of 2250 psia

The methodology used for the supplemental input to dose analysis for NOP/NOT conditions is the same as that described in Reference 1.

5.3.3.3 Acceptance Criteria

The current licensing basis analyses were performed to calculate mass transfer data for input to the radiological consequences analyses to determine if the licensing basis SGTR input to dose analysis is bounding. As such, the results of the supplemental SGTR input to dose analysis at NOP/NOT conditions must be bounded by the current licensing basis SGTR input to dose analysis in order for the SGTR doses presented in UFSAR Chapter 14.2.4.5 to remain valid.

5.3.3.4 Description of Analyses and Evaluations

A sensitivity calculation was performed on the supplemental SGTR input to dose analysis using the NOP/NOT T_{avg} of 571.0°F and RCS pressure of 2250 psia. The sensitivity followed the methodology of Reference 1.

5.3.3.5 Results

The mass releases from the supplemental SGTR input to dose analysis remain bounded by the current licensing basis SGTR input to dose analysis presented in UFSAR Chapter 14.2.4.

5.3.3.6 Conclusions

The supplemental input to dose analysis continues to confirm that the current licensing basis SGTR input to dose analysis remains bounding at NOP/NOT conditions. As such, the SGTR doses presented in UFSAR Chapter 14.2.4.5 remain valid.

5.3.3.7 References

1. C0801-02, "Donald C. Cook Nuclear Plant Units 1 and 2 Final Response to Nuclear Regulatory Commission Request for Additional Information Regarding License Amendment Request for Control Room Habitability (TAC Nos. MA9394 and MA9395)," dated August 17, 2001. (Available in NRC ADAMS under Accession Number ML012330380)

5.3.4 SGTR – Supplemental Margin-to-Overfill

5.3.4.1 Introduction and Background

The SGTR MTO analysis demonstrates that a ruptured SG tube does not result in overfill into the main steam piping during the accident. This analysis supports the SGTR radiological dose analysis assumption that liquid water is not released from the main steam safety or power-operated relief valves (PORVs).

The SGTR MTO analysis that supports the UFSAR for Cook Unit 1 is described in References 1 and 2 and was approved by the USNRC in Reference 3.

5.3.4.2 Input Parameters and Assumptions

The SGTR MTO analysis was evaluated using the following NOP/NOT conditions:

1. T_{avg} of 571.0°F
2. RCS pressure of 2250 psia

The methodology used for the MTO analysis for NOP/NOT conditions is the same as that described in References 1 and 2 and approved by the USNRC in Reference 3, with conservative changes to address the unrelated concerns raised in NSAL-07-11 (Reference 4).

5.3.4.3 Acceptance Criteria

The current licensing basis analyses were performed to determine if the secondary side of the ruptured SG would completely fill with water. The available secondary side volume of a single SG is 5549.7 ft^3. Margin to overfill is demonstrated, provided the peak SG secondary side water volume is less than this value. No credit is taken for the volume of the nozzle or any steam piping.

5.3.4.4 Description of Analyses and Evaluations

The impact of the increased pressure and temperature on the SGTR MTO analysis that supports the UFSAR for Cook Unit 1 were examined. Historical sensitivity studies on the Cook Unit 1 SGTR MTO analysis were reviewed for the impact of returning to a RCS pressure of 2250 psia and raising the T_{avg} to 571.0°F. These historical sensitivities demonstrate that both the increase in RCS pressure and the increase in T_{avg} result in increased margin to SG overfill. The increased margin is due to increased steaming from the ruptured SG. With an increase in T_{avg}, the secondary pressure is higher such that the secondary relief valve lifts earlier. With an increase in the RCS pressure, safety injection (SI) actuation on low pressurizer pressure is delayed. The resulting delay in cold SI flow to the RCS increases energy transfer to the SG secondary side. Thus, these historical sensitivities are sufficient for evaluating the impact of the Return to NOP/NOT Program in and of themselves and do not require a reanalysis of the SGTR MTO.

The Cook Unit 1 SGTR MTO analysis was, however, updated to reflect NOP/NOT input conditions and to include plant specific sensitivity calculations to determine the conservative direction for the assumed decay heat, auxiliary feedwater (AFW) enthalpy and SI enthalpy to address the concerns identified in NSAL-07-11 (Reference 4).

5.3.4.5 Results

Historical sensitivities have shown that an increase in RCS pressure and T_{avg} is a benefit for the Cook Unit 1 SGTR MTO analysis. These sensitivities show the Return to RCS NOP/NOT Program is a benefit for the SGTR MTO analysis.

The Cook Unit 1 SGTR MTO analysis was updated to reflect NOP/NOT input conditions and to include plant specific sensitivity calculations to determine the conservative direction for the assumed decay heat, AFW enthalpy and SI enthalpy. The analysis determined that low decay heat (ANS 1979-2σ) produced more limiting results than the high decay heat (ANS 1971+20%) considered in Reference 1, and confirmed that maximum AFW and SI enthalpies (as modeled in Reference 1) produced more limiting results. The limiting case resulted in a peak ruptured SG water volume of approximately 5480 ft^3, leaving approximately 69 ft^3 of MTO.

5.3.4.6 Conclusions

The SGTR MTO analysis, using the methodology of Reference 1 modified to address the concerns of NSAL-07-11 (Reference 4), shows margin at NOP/NOT conditions. The SGTR MTO analysis demonstrates that the ruptured SG does not overfill into the main steam piping and supports the SGTR dose assumption that liquid water is not released through the main steam relief valves.

5.3.4.7 References

1. C1000-11, "Donald C. Cook Nuclear Plant Units 1 and 2 License Amendment for Changes in Steam Generator Tube Rupture Analysis Methodology," dated October 24, 2000. (Available in NRC ADAMS under Accession Number ML003762982)

2. C0601-21, "Donald C. Cook Nuclear Plant Units 1 and 2 Response to Nuclear Regulatory Commission Request for Additional Information Regarding License Amendment for Changes in Steam Generator Tube Rupture Analysis Methodology (TAC Nos. MB0739 and MB0740)," dated June 29, 2001. (Available in NRC ADAMS under Accession Number ML011860097)

3. Letter from John F. Stang (U.S. NRC) to Robert P. Powers (AEP), "Donald C. Cook Nuclear Plant, Units 1 and 2 – Issuance of Amendments (TAC Nos. MB0739 and MB0740)," dated October 21, 2001. (Available in NRC ADAMS under Accession Number ML012690136)

4. NSAL-07-11, "Decay Heat Assumption in Steam Generator Tube Rupture Margin-to-Overfill Analysis Methodology," dated November 15, 2007.

5.4 LOCA MASS & ENERGY RELEASES AND CONTAINMENT RESPONSE

5.4.1 Short-Term LOCA Mass and Energy Releases and Containment Response

5.4.1.1 Introduction and Background

Short-term pressure pulses resulting from high energy line breaks create the situation where a high mass flux allows local pressure to build up at a rate faster than the overall containment pressure, leading to a challenge to structural integrity within containment. Containment subcompartments, such as the pressurizer enclosure, are designed to withstand the effects of a short-term mass and energy (M&E) release transient, and as such, must be re-evaluated due to changes in operating conditions. For ice condenser containments, the subcompartments of interest relative to LOCA M&E are the following; pressurizer enclosure, loop subcompartments, and the upper and lower reactor cavity. The latest short-term containment analysis will be evaluated relative to current containment design information.

5.4.1.2 Input Parameters and Assumptions

The following NOP/NOT data were used as key input to the evaluation.

1. Vessel/core inlet temperature of 514.1°F, including reduction due to 5.1°F uncertainty
2. Reactor coolant system (RCS) pressure is 2317 psia, including 67 psi uncertainty
3. Core power is 3315 MWt, including 0.34 percent calorimetric power uncertainty

5.4.1.3 Acceptance Criteria

There are three short-term subcompartment response analyses performed; pressurizer enclosure, loop subcompartments, and upper and lower reactor cavity. In order for the results of these current analyses to be acceptable, the maximum calculated differential pressure across containment structures cannot exceed the design pressure of the particular element (found in Table 5.2-8 of the UFSAR). If the RCS conditions at NOP/NOT are found to be bounded by those previously analyzed, the current M&E releases will remain acceptable.

5.4.1.4 Description of Analyses and Evaluations

Short-term LOCA M&E releases, which are used as input to the subcompartment analyses, are sensitive to two parameters; RCS temperature and RCS pressure. The RCS operating conditions at NOP/NOT conditions will be compared with those from the analysis of record (AOR) to determine if the AOR remains bounding.

There were no new calculations made to support short-term LOCA M&E releases and containment response. The nuclear steam supply system (NSSS) design parameters which support operation of Cook Unit 1 at NOP/NOT conditions are in subsection 5.4.1.2. These new conditions were compared with the data which supports the short-term LOCA M&E and containment response AOR.

5.4.1.5 Results

The limiting breaks in the AOR for short-term M&E release and containment response are bounding relative to a vessel/core inlet temperature of 506.6°F (AOR NSSS design core/vessel inlet temperature of 511.7°F minus 5.1°F uncertainty) and a pressure of 2317 psia (AOR NSSS design RCS pressure of 2250 psia plus 67 psi uncertainty). The NSSS design parameters in subsection 5.4.1.2 show a minimum vessel/core inlet temperature of 514.1°F (including uncertainty) and pressure of 2317 psia (including uncertainty). The temperature and pressure conditions of NOP/NOT are bounded and no new short term M&E releases will be required.

5.4.1.6 Conclusions

Because the NOP/NOT RCS temperature and pressure are bounded by the current analyses, it is concluded that these analyses will support Cook Unit 1 NOP/NOT conditions.

5.4.2 Long-Term LOCA Mass and Energy Releases and Containment Response

5.4.2.1 Introduction and Background

The purpose of this evaluation was to determine if changes related to Cook Unit 1 operation at NOP/NOT impact the LOCA M&E and containment integrity analyses. The purpose of the LOCA M&E release analysis is to demonstrate the ability of the containment safeguards systems to mitigate the consequences of a hypothetical large-break LOCA. The methodology used to perform this analysis is documented in Reference 1. The containment analysis methodology is documented in Reference 2.

5.4.2.2 Input Parameters and Assumptions

The following NOP/NOT data were used as key input to this evaluation:

1. Vessel inlet temperature of 543.6°F
2. Vessel outlet temperature of 609.1°F
3. Core inlet temperature of 543.6°F
4. Steam generator (SG) secondary pressure of 870 psia
5. SG secondary temperature of 527.9°F

6. Core stored energy of 4.76 full power seconds
7. Low pressurizer pressure reactor trip nominal setpoint increase from 1875 to 1950 psig
8. Containment spray (315 seconds) and containment air recirculation fan (300 seconds) actuation post accident

There are no other NOP/NOT related items which would impact the LOCA M&E and containment integrity analyses.

5.4.2.3 Acceptance Criteria

The Cook Unit 1 containment design pressure is 12 psig, and calculated post LOCA pressures must remain below this limit. The acceptability of this evaluation was based upon determining whether or not the current Cook Unit 1 LOCA M&E and containment integrity analysis remained bounding relative to operation at NOP/NOT, including the effects of delayed containment spray (CTS) and containment air recirculation fan operation.

5.4.2.4 Description of Analyses and Evaluations

There was no new LOCA M&E analysis performed for the NOP/NOT scope. The scope of work was limited to an evaluation of the current LOCA M&E AOR. The evaluation is to determine if the current LOCA M&E and containment integrity AOR supports operation for Cook Unit 1 at NOP/NOT. For the LOCA M&E analysis, this was done by comparing the Cook Unit 1 NOP/NOT conditions to the inputs assumed in the current LOCA M&E AOR. For the containment integrity analysis, this was completed by evaluating the effects of increased delay times for CTS actuation and containment air recirculation fan actuation on the LOCA containment integrity analysis.

5.4.2.5 Results

The key parameters assumed in the LOCA M&E AOR are compared below with the parameters developed to support Cook Unit 1 NOP/NOT. In all cases, the AOR values remain bounding. The RCS temperatures below for AOR and NOP/NOT both include uncertainties of 5.1°F. The secondary pressure and temperature values are the high safety analysis values provided with the NSSS design parameters. Core stored energy is provided as a bounding value, and no additional uncertainty is applied in the LOCA M&E analysis.

Parameter	AOR	NOP/NOT
Vessel Inlet Temperature (°F)	552.5	548.7
Vessel Outlet Temperature (°F)	620.3	614.2
Core Inlet Temperature (°F)	552.5	548.7
SG Secondary Pressure (psia)	878	870
SG Secondary Temperature (°F)	529.1	527.9
Core Stored Energy (full power seconds)	4.95	4.76

The evaluation indicated that the effect of the containment air recirculation fan delay as well as the CTS delay resulted in a pressure increase of 0.0300 psi. The AOR peak containment pressure is 11.6884 psig and the combined effect of CTS and containment air recirculation fan delays results in a peak containment pressure of 11.7184 psig.

5.4.2.6 Conclusions

Subsection 5.4.2.5 indicates that the NSSS design parameters supporting the Cook Unit 1 LOCA M&E AOR are bounding relative to the NOP/NOT NSSS design parameters. It is concluded that the NSSS design parameter inputs are conservative and operation at NOP/NOT is supported.

The core stored energy in the Cook Unit 1 LOCA M&E AOR, 4.95 full power seconds, is greater than the value calculated for NOP/NOT (including effects of thermal conductivity degradation) of 4.76 full power seconds. It was concluded that the fuel modeling in the AOR supports operation at NOP/NOT.

There was no impact due to an increase in the low pressurizer pressure reactor trip nominal setpoint since rod drop is not credited in the LOCA M&E release calculations (the core shuts down due to voiding). It is concluded that the AOR calculations for Cook Unit 1 LOCA M&E support operation at NOP/NOT.

Sensitivity studies showed that the effect of the CTS and containment air recirculation fan actuation delays was a containment peak pressure increase to 11.7184 psig which is acceptable for containment integrity because the containment design limit of 12 psig is not exceeded.

The evaluation of the long-term LOCA M&E and peak containment pressure is predicated upon the continued application of the operability assessment supporting NSAL-11-5 (Reference 3), in conjunction with the AOR.

5.4.2.7 References

1. Westinghouse Topical Report WCAP-10326-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version," May, 1983.

2. Westinghouse Topical Report WCAP-8355-A, "Long Term Ice Condenser Code – LOTIC Code," April, 1976.

3. Westinghouse Nuclear Safety Advisory Letter NSAL-11-5, "Westinghouse LOCA Mass and Energy Release Calculation Issues," July 25, 2011.

5.5 MAIN STEAMLINE BREAK MASS & ENERGY RELEASES AND CONTAINMENT RESPONSE

Steam line ruptures inside containment result in significant releases of high-energy fluid to the containment environment, producing elevated containment temperatures and pressures. The magnitude of the releases is dependent upon the plant's initial operating conditions and the size of the rupture, as well as the configuration of the plant's steam system and the containment design. To ensure that the worst cases for either containment temperature or pressure are identified, analyses consider a variety of postulated pipe breaks, which encompass wide variations in plant operation, safety system performance, and break size, to determine the most challenging main steamline break (MSLB) mass and energy (M&E) releases and containment response.

Section 5.5 is divided into two subsections that describe the MSLB M&E release inside containment (subsection 5.5.1) and the associated containment integrity analysis (subsection 5.5.2) performed for the Cook Unit 1 Return to Reactor Coolant System (RCS) Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) Program. The general approach taken for NOP/NOT was to replace the current bounding analyses of record (AOR) for MSLB M&E containment by preparing Cook Unit 1-specific analyses that reflect the current nuclear fuel design, replacement steam generator (RSG) design, and the revised NOP/NOT operating conditions. In addition, existing USNRC-approved methodologies for MSLB M&E containment analyses were used in the new Cook Unit 1 NOP/NOT analyses.

5.5.1 Main Steamline Break Mass & Energy Releases Inside Containment

5.5.1.1 Introduction and Background

The AOR for MSLB M&E releases inside containment for Cook Unit 1 was performed using bounding analysis inputs that include an assumed nuclear steam supply system (NSSS) power of 3600 MWt. The most recent evaluation of the AOR specific to Cook Unit 1 was documented as part of the Cook Unit 1 measurement uncertainty recapture (MUR) program. This evaluation included an analysis of a limited number of MSLB cases with RSGs to support the NSSS power increase associated with the MUR.

Since there is no Cook Unit 1-specific MSLB M&E analysis that includes a full steamline break spectrum with Model Babcock & Wilcox International (BWI) - Series 51 RSGs and the currently licensed NSSS power level, a decision was made to prepare such an analysis for the Return to RCS NOP/NOT Program. The scope presented in subsection 5.5.1 is a full-spectrum analysis of the MSLB inside containment, producing M&E releases for input to the containment pressure and temperature response analysis that is described in subsection 5.5.2. The calculated MSLB M&E releases documented for the Return to RCS NOP/NOT Program supersede the prior AOR for Cook Unit 1.

5.5.1.2 Input Parameters and Assumptions

The analysis inputs, assumptions, and methods pertaining to the MSLB M&E releases inside containment are based on the approved methodologies documented in References 1 and 2. Input considerations include break size, existence of water entrainment (i.e., steam quality < 1.0) in the steam effluent, and the type of protection signal actuated for split breaks. [Note that the USNRC Acceptance Letter and Safety Evaluation in Sections A and B of Reference 2 apply to both References 1 and 2.]

To determine the effects of plant power level and break area on the M&E releases from a ruptured steamline, spectra of both variables have been evaluated. At plant power levels of 100.34, 70, 30, and 0 percent of nominal full-load NSSS power, the following break sizes have been defined.

- A full double-ended rupture (DER) downstream of the SG outlet nozzle integral flow restrictor in one steamline. Note that a DER is defined as a rupture in which the steam pipe is completely severed and the ends of the break fully displace from each other. Saturated steam is released from the DER at each initial power level. Since the full DER represents the largest break of the main steamline and produces the highest mass flowrate from the faulted-loop SG, smaller DER break sizes are not generally considered. The exception is listed in the next bullet.

- A small DER with dry saturated steam release is analyzed at 0 percent initial power. The small DER under these conditions is more limiting than a full DER with entrained liquid in the break effluent.

- A small split rupture, the largest break that will neither generate a steamline isolation signal from the primary protection equipment nor result in water entrainment in the break effluent. Reactor protection and safety injection (SI) actuation functions are obtained from containment pressure signals.

The 18 cases included in the Cook Unit 1 analysis for the Return to RCS NOP/NOT Program have been chosen based on the selection of similar steamline ruptures that have been analyzed to support the results presented in the D. C. Cook Nuclear Plant Updated Final Safety Analysis Report (UFSAR), subsection 14.3.4.4. The cases, listed in subsection 5.5.1.4 of this report, have been analyzed at the currently licensed NSSS power for Cook Unit 1 and reflect design changes associated with the replacement Model BWI-Series 51 SGs and the 15x15 Upgrade Fuel. Other assumptions regarding important plant conditions and features are discussed in the following bullets.

- Piping discharge resistances are not included in the calculation of the releases resulting from the steamline ruptures (Moody Curve for an $f(\ell/D) = 0$ is used). This is consistent with the expectations of the USNRC as presented in subsection 6.2.1.4 of the Standard Review Plan.

- Although entrainment is expected for all DER cases, dry saturated steam (no entrainment) in the break effluent is initially, conservatively assumed. In the event that the containment temperature response exceeds the temperature criterion in subsection 5.5.2., the option in References 1 and 2 to model water entrainment in the steam, which effectively reduces the energy content of the steam released into containment from the faulted SG, is considered.* For DER cases, the water entrainment option maintains analytical conservatism in the analysis. Consistent with USNRC-approved References 1 and 2, the NOP/NOT analysis includes an uncertainty of 0.1 added to the quality of the break effluent whenever water entrainment is modeled in the steam release. Also, when assumed, entrainment in the break effluent is from only the SG in the faulted loop.

  * [It is noted that the entrainment models in References 1 and 2 were based on Westinghouse Model D and Model 51 SG designs. The existence of water entrainment in the steam release during a DER is, however, independent of the SG design since the high steam flow associated

with the DER greatly exceeds the capacity of any moisture separator equipment, thereby making the separators ineffective in preventing water entrainment in the exiting steam. The impact of a DER on SG moisture separators is specifically documented in References 1 and 2. The conclusion that the entrainment model in References 1 and 2 can be applied to SG designs other than the original Westinghouse Model D and Model 51 during a postulated MSLB DER is documented in References 3 and 4. Collectively, these factors support the application of the Reference 1 and 2 water entrainment model to the Cook Unit 1 Model BWI-Series 51 SGs.]

- The contribution from the secondary plant steam piping is included in the M&E release calculations. The flowrate is determined using the Moody correlation, the pipe cross-sectional area, and the initial steam pressure. This blowdown is calculated only for the full DER steamline break. A conservative steam piping volume representing the main steam piping from the four SGs up to the inlet to the turbine is used in this blowdown calculation.

- For the split-rupture steamline break, the unisolable steam mass in the piping is included as part of the initial inventory in the faulted-loop SG since the break is not large enough to cause a sudden decompression of the piping.

5.5.1.3 Acceptance Criteria

The acceptance criteria associated with the MSLB event resulting in M&E release inside containment are based on an analysis that provides sufficient conservatism to show that the containment design margin is maintained. The specific criteria applicable to this analysis are related to the assumptions regarding power level, stored energy, the break flow model, main and auxiliary feedwater (AFW) flow, steamline and feedwater isolation, and single failure such that the calculated containment peak temperature and pressure are maximized. These assumptions have been included in this MSLB M&E release analysis, as discussed in Reference 1.

5.5.1.4 Description of Analyses and Evaluations

The NOP/NOT analysis for MSLB M&E releases inside containment is based on the approved methodologies in References 1 and 2 and the inputs and assumptions listed in subsection 5.5.1.2.

The following cases of the MSLB inside containment have been analyzed at the noted conditions for the Cook Unit 1 Return to RCS NOP/NOT Program.

- Full DER (1.4 ft^2) at 100.34 percent power – main steam isolation valve (MSIV) and main feedwater isolation valve (MFIV) single failures

- Full DER (1.4 ft^2) at 100.34 percent power – AFW runout protection and MFIV single failures

- Split rupture (0.865 ft^2) at 100.34 percent power – MSIV and MFIV single failures

- Split rupture (0.865 ft^2) at 100.34 percent power – AFW runout protection and MFIV single failures

- Full DER (1.4 ft^2) at 70 percent power – MSIV and MFIV single failures
- Full DER (1.4 ft^2) at 70 percent power – AFW runout protection and MFIV single failures
- Split rupture (0.857 ft^2) at 70 percent power – MSIV and MFIV single failures
- Split rupture (0.857 ft^2) at 70 percent power – AFW runout protection and MFIV single failures
- Full DER (1.4 ft^2) at 30 percent power – MSIV and MFIV single failures
- Full DER (1.4 ft^2) at 30 percent power – AFW runout protection and MFIV single failures
- Split rupture (0.834 ft^2) at 30 percent power – MSIV and MFIV single failures
- Split rupture (0.834 ft^2) at 30 percent power – AFW runout protection and MFIV single failures
- Full DER (1.4 ft^2) at 0 percent power, entrainment in the faulted-loop SG – MSIV and MFIV single failures
- Full DER (1.4 ft^2) at 0 percent power, entrainment in the faulted-loop SG – AFW runout protection and MFIV single failures
- Small DER (1.0 ft^2) at 0 percent power – MSIV and MFIV single failures
- Small DER (1.0 ft^2) at 0 percent power – AFW runout protection and MFIV single failures
- Split rupture (0.808 ft^2) at 0 percent power – MSIV and MFIV single failures
- Split rupture (0.808 ft^2) at 0 percent power – AFW runout protection and MFIV single failures

5.5.1.5 Results

Using the MSLB analysis methodologies documented in References 1 and 2 as the bases, including Cook Unit 1 plant-specific parameters, the M&E release rates for each of the steamline break cases noted in subsection 5.5.1.4 have been developed for use in containment temperature and pressure response analyses. All of the analyzed breaks conservatively assumed dry saturated steam releases (no entrainment) except the full DER at 0 percent initial power. As a result, the small DER with dry saturated steam release was analyzed at 0 percent power, represented by a 1.0 ft^2 break (smaller than the area of a single integral flow restrictor) from the faulted-loop SG and a 1.0 ft^2 break for the reverse-flow blowdown from the intact-loop SGs. Table 5.5.1-1 provides the sequence of events for each of the 18 steamline break sizes analyzed for Cook Unit 1 for the Return to RCS NOP/NOT Program.

For the full DER MSLB at all power levels, the first protection system signals actuated are low steamline pressure (lead/lag compensated, required in 2 of the 4 loops) that initiates steamline isolation and actuates the 'S' signal; the 'S' signal produces a reactor trip signal and actuates SI. Feedwater system isolation and AFW actuation also occur as a result of the 'S' signal.

For the split-rupture steamline breaks at all power levels, no mitigation signal is received from any secondary-side signal produced by the primary protection equipment. The first protection system signals actuated are assumed to be the high-1 containment pressure, which initiates the 'S' signal; the 'S' signal produces a reactor trip signal and actuates SI. Feedwater system isolation and AFW actuation also occur as a result of the 'S' signal. Steamline isolation is initiated following receipt of the high-2 containment pressure signal.

For the small DER MSLB at 0 percent power, the coincidence logic for the low steamline pressure signals in 2 of the 4 loops is not satisfied and no other mitigation signal is received from any secondary-side signal. Credit is taken for the high-1 containment pressure signal and the high-2 containment pressure signal as discussed in the paragraph above for the split-rupture steamline breaks.

5.5.1.6 Conclusions

The M&E releases from the 18 MSLB cases inside containment have been analyzed for Cook Unit 1 to support the Return to RCS NOP/NOT Program. The MSLB M&E releases discussed in this report have been provided for use in the containment response analysis for Cook Unit 1.

5.5.1.7 References

1. WCAP-8822 (Proprietary) and WCAP-8860 (Nonproprietary), "Mass and Energy Releases Following a Steam Line Rupture," September 1976; WCAP-8822-S1-P-A (Proprietary) and WCAP-8860-S1-A (Nonproprietary), "Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986.

2. WCAP-8821-P-A (Proprietary) and WCAP-8859-A (Nonproprietary), "TRANFLO Steam Generator Code Description," June 2001.

3. USNRC letter (Deirdre W. Spaulding) to NMC (Fred J. Cayia), "Point Beach Nuclear Plant Units 1 and 2 – Issuance of Amendments RE: Change of Containment Maximum Pressure Technical Specification Limit (TAC NOS. MB3870 and MB3871)," November 26, 2002. (archived in EDMS as an attachment to Reference 27) [contains the USNRC SER: Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 206 to Facility Operating License No. DPR-24 and Amendment No. 211 to Facility Operating License No. DPR-27, Nuclear Management Company, LLC, Point Beach Nuclear Plant, Units 1 and 2, Docket Nos. 50-266 and 50-301]

4. USNRC letter (Thomas J. Wengert) to Northern States Power (Michael D. Wadley), "Prairie Island Nuclear Generating Plant, Units 1 and 2 - Issuance of Amendments RE: Technical Specifications Changes to Allow Use of Westinghouse 0.422-inch OD 14x14 Vantage+ Fuel (TAC NOS. MD9142 and MD9143)," July 1, 2009. (ADAMS Accession No. ML0914/ML091460809) [contains the USNRC SER: Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 192 to Facility Operating License No. DPR-42 and Amendment No. 181 to Facility Operating License No. DPR-60, Northern States Power Company - Minnesota, Prairie Island Nuclear Generating Plant, Units 1 and 2, Docket Nos. 50-282 and 50-306]

Table 5.5.1-1 Transient Summary for the Spectrum of MSLB M&E Releases Inside Containment

Initial Power, Single Failure	Break Type	Reactor Trip Signal	Rod Motion (sec)	AFW Initiation/ Termination (sec)	Main Feedwater Isolation (sec)	Steamline Isolation (sec)[1,2]	End of Steam Mass Release (sec)
100.34%, MSIV/MFIV	Full DER	LSP-SI	4.72	1.72/1800	45.72	12.72	1803
100.34%, AFW/MFIV	Full DER	LSP-SI	4.72	1.72/1800	45.72	12.72	1808
100.34%, MSIV/MFIV	Split	High-1	5.3	2.3/1800	46.3	17.5	1808
100.34%, AFW/MFIV	Split	High-1	5.3	2.3/1800	46.3	17.5	1816
70%, MSIV/MFIV	Full DER	LSP-SI	4.39	1.39/1800	45.39	12.39	1803
70%, AFW/MFIV	Full DER	LSP-SI	4.39	1.39/1800	45.39	12.39	1808
70%, MSIV/MFIV	Split	High-1	5.2	2.2/1800	46.2	17.3	1809
70%, AFW/MFIV	Split	High-1	5.2	2.2/1800	46.2	17.3	1817
30%, MSIV/MFIV	Full DER	LSP-SI	4.21	1.21/1800	45.21	12.21	1803
30%, AFW/MFIV	Full DER	LSP-SI	4.21	1.21/1800	45.21	12.21	1810
30%, MSIV/MFIV	Split	High-1	5.1	2.1/1800	46.1	17.0	1811
30%, AFW/MFIV	Split	High-1	5.1	2.1/1800	46.1	17.0	1828
0%, MSIV/MFIV	Full DER	LSP-SI	4.28	0.0/1800	45.28	12.28	1803
0%, AFW/MFIV	Full DER	LSP-SI	4.28	0.0/1800	45.28	12.28	1807
0%, MSIV/MFIV	Small DER	High-1	3.9	0.0/1800	44.9	13.4	1805
0%, AFW/MFIV	Small DER	High-1	3.9	0.0/1800	44.9	13.4	1810

Table 5.5.1-1 Transient Summary for the Spectrum of MSLB M&E Releases Inside Containment (cont.)

Initial Power, Single Failure	Break Type	Reactor Trip Signal	Rod Motion (sec)	AFW Initiation/ Termination (sec)	Main Feedwater Isolation (sec)	Steamline Isolation (sec)(1,2)	End of Steam Mass Release (sec)
0%, MSIV/MFIV	Split	High-1	5.2	0.0/1800	46.2	17.2	1808
0%, AFW/MFIV	Split	High-1	5.2	0.0/1800	46.2	17.2	1814

Key LSP – low steam pressure
SI – safety injection
MSIV – main steam isolation valve
High-1 – containment high-1 pressure

(1) For the MSIV-failure cases, steamline isolation occurs only in the 3 unfaulted steamlines; there is no closure of the MSIV in the faulted steamline.

(2) For the split breaks, the signal for steamline isolation is containment high-2 pressure.

5.5.2 Main Steamline Break Containment Response

5.5.2.1 Introduction and Background

The containment integrity analysis for Cook Unit 1 is performed to verify that the maximum containment pressure and temperature acceptance criteria are not exceeded during postulated M&E releases inside containment. Historically, the containment pressure limits for the Cook Plant from M&E releases are based on the RCS LOCA double-ended pump suction break, while the containment temperature limit is most severely challenged by a postulated MSLB in the lower compartment.

The current AOR for the MSLB containment response for Cook Unit 1 was performed using bounding analysis inputs that include an assumed NSSS power of 3600 MWt. The most recent evaluation of the AOR specific to Cook Unit 1 was documented as part of the MUR program. This evaluation included an analysis of a limited number of MSLB cases with RSGs to support the NSSS power increase associated with the MUR. This analysis is described in Attachment 3 of the MUR License Amendment Request (Reference 1) and accepted in Reference 2.

Since there is no Cook Unit 1-specific MSLB containment analysis that includes a full steamline break spectrum with Model BWI-Series 51 RSGs and the currently licensed NSSS power, a decision was made to prepare such an analysis for the Return to RCS NOP/NOT Program. The scope presented in subsection 5.5.2 is a full-spectrum analysis of the Cook Unit 1 MSLB containment pressure and temperature responses. The calculated MSLB containment pressure and temperature responses documented for the Return to RCS NOP/NOT Program supersede the prior analyses for Cook Unit 1.

5.5.2.2 Input Parameters and Assumptions

The containment response to an MSLB is dependent on containment configuration, operating conditions prior to steamline failure, break size, and the single failure assumed. The major input assumptions used in the MSLB containment response analysis for the Cook Unit 1 Return to RCS NOP/NOT Program include the following.

- Minimum containment safeguards are employed consistent with the single failure of an emergency diesel generator (EDG) following a loss of offsite power (LOOP). The minimum containment heat removal capability is provided by one-of-two spray pumps and one-of-two containment air recirculation fans.

- The protection systems available to mitigate the effects of an MSLB inside containment include high-1 containment pressure having a setpoint of 1.7 psig, and high-2 containment pressure having a setpoint of 3.5 psig. The high-1 signal actuates the 'S' signal, which produces a reactor trip signal, actuates SI and main feedwater isolation, and starts the containment air recirculation fans. The high-2 signal actuates steamline isolation and starts the containment spray (CTS) pumps.

- A CTS pump flow of 1,960 gpm is used in the upper compartment and 706 gpm is assumed in the lower compartment. The diesel loading sequence for the CTSs to energize and come up to full flow is 315 seconds after reaching the high-2 containment setpoint.

- The containment air recirculation fan is effective 300 seconds after the high-1 containment pressure signal is actuated. The volumetric flowrate is 39,000 cfm emptying air from the upper compartment to the lower compartment.

- A total initial ice mass of 2.2 x10^6 lbs. The MSLB event does not melt the entire initial ice mass modeled during the time frame of the transient analyzed.

- A uniform distribution of steam flow into the ice bed.

- The initial conditions in the containment are a temperature of 120°F in the lower compartment, 120°F in the dead-ended compartment, a temperature of 57°F in the upper compartment, and a temperature of 27°F in the ice condenser. All containment volumes are at a pressure of 0.3 psig and a relative humidity of 15 percent, with the exception of the ice bed, which is 100 percent relative humidity for all cases.

- The refueling water storage tank (RWST) temperature is 105°F.

- The spurious operation of the upper containment ventilation heaters is included in the model as a 288 kw additional heat input.

- The heat transfer coefficients to the containment structures are based on the work of Tagami. An explanation of their manner of application is given in Reference 4.

- No revaporization of the condensation inside containment is assumed consistent with the licensing conditions stipulated in Reference 5.

- Containment structural heat sinks based on the information presented in the D. C. Cook Nuclear Plant UFSAR Table 14.3.4-4 were used.

- The material property data for the containment structural heat sinks based on the information presented the D. C. Cook Nuclear Plant UFSAR Table 14.3.4-5 were used.

- The MSLB M&E release rates discussed in subsection 5.5.1 are specific to Cook Unit 1 at an NSSS power of 3327 MWt, with RSGs, and 15x15 Upgrade Fuel.

5.5.2.3 Acceptance Criteria

The acceptance criteria associated with the analysis of the containment response attributed to MSLB M&E releases relate to the design pressure and the transient temperature inside containment. The following containment design limits from the Cook Plant UFSAR reflect the acceptance criteria for the MSLB containment analysis documented for the Cook Unit 1 Return to RCS NOP/NOT Program.

- Containment design pressure is 12 psig
- Containment peak transient temperature limit is 324.7°F

5.5.2.3 Description of Analyses and Evaluations

Consistent with the current AOR for Cook Unit 1 MSLB M&E Containment, the containment pressure and temperature response transient for NOP/NOT conditions has been analyzed using the LOTIC-3 (Reference 5) computer code, which was developed to analyze steamline breaks in an ice condenser plant. Details of the LOTIC computer code are described in References 3 through 5. The LOTIC-3 computer model has been found to be acceptable for the analysis of the design-basis MSLB event, as documented in Reference 5.

Three conditions exist in the USNRC's Safety Evaluation of Reference 5 before LOTIC-3 can be used for MSLB containment analysis. The conditions are:

1. M&E releases input to LOTIC-3 from the steam system must be calculated with a model approved by the USNRC.

2. LOTIC-3 Option 2, which assumes steam to be condensed and added to the sump from heat transfer to the structural heat sinks, must be used for break sizes producing no liquid entrainment and for all break sizes until liquid entrainment models are approved.

3. A break spectrum analysis is required for each plant to demonstrate that the most severe containment conditions have been identified.

Subsection 5.5.1 documents that the MSLB M&E releases have been calculated with models approved by the USNRC, thus satisfying the first condition in the USNRC Safety Evaluation for LOTIC-3. The models used for the MSLB M&E analysis for NOP/NOT consider steam releases with and without entrainment, as described in subsection 5.5.1. The second Safety Evaluation condition is met in the NOP/NOT analysis by use of the conservative 0 percent condensate revaporization and convective heat flux models (Option 2), regardless of whether entrainment was modeled in MSLB calculations. As discussed in subsection 5.5.1.4, entrainment has been assumed for the 0 percent power full DER cases. Finally, Safety Evaluation Condition 3 is met in the new NOP/NOT MSLB M&E analysis by the inclusion of a break spectrum that includes full DERs, a small DER, and split breaks. This spectrum ensures that the most severe containment temperature response has been analyzed.

Eighteen licensing-basis MSLB M&E release cases have been analyzed, as delineated in subsection 5.5.1.4. For each of the MSLB cases, the containment response has been analyzed for the Cook Unit 1 Return to RCS NOP/NOT Program to calculate the transient pressures and temperatures in the upper and lower compartments of the ice condenser containment. Tables of transient M&E release data have been included as input to the LOTIC-3 model for each of the eighteen MSLB cases analyzed. For the full DER cases, the reverse steam flow blowdown M&E releases have also been included as input to the LOTIC-3 model.

5.5.2.4 Results

Using the containment analysis methodology for MSLBs documented in Reference 5 as a basis and the Cook Unit 1-specific design input parameters, the containment pressure and temperature transient responses for each of the steamline break cases noted in subsection 5.5.1.4 have been calculated.

Table 5.5.2-1 provides the peak containment pressure and lower compartment temperature for each of the steamline break sizes analyzed for Cook Unit 1 for the Return to RCS NOP/NOT Program.

The maximum peak containment pressure occurs following the 0 percent power full DER MSLBs based on the initial blowdown of the main steam piping and the faulted SG. The peak pressure is 9.72 psig, which is less than the pressure criterion of 12 psig. The containment pressure transient is shown in Figure 5.5.2-1. In general, the DER MSLBs produce a higher peak pressure than do the split breaks due to the rapid steam release from the larger break area until the time of steamline isolation. The peak pressures resulting from the full DERs occur at the termination of the reverse blowdown or the time of steamline isolation.

The peak lower compartment temperature occurs following the 100.34 and 70 percent power split breaks with an MSIV single failure. The peak temperatures are 324.67°F and 324.66°F respectively, which are less than the temperature criterion of 324.7°F. The upper and lower compartment temperature transients for the 100.34 percent power split break case are shown in Figure 5.5.2-2. In general, the split breaks produce a higher temperature for a longer duration than do the full DERs (which result in a peak at the time of steamline isolation) due to the longer time required to empty the contents of the faulted SG.

The 0 percent power full 1.4-ft^2 DER has included entrained liquid within the steam released from the faulted SG to reduce the peak lower compartment temperature at the time of steamline isolation due to the reduction in the steam enthalpy. A small 1.0-ft^2 DER has been analyzed to show the effect of dry saturated steam (no entrainment) on a break size less than the full DER. The effect of the small DER is to extend the duration of the total steam energy released from the break and confirm that the small DER is not a limiting break.

The peak containment temperatures for the NOP/NOT MSLB cases with dry saturated steam are in the range of 323.71°F to 324.67°F, all less than the temperature acceptance criterion.

5.5.2.5 Conclusions

The containment pressure and temperature transient responses for each of the 18 steamline break cases have been analyzed for Cook Unit 1 to support the Return to RCS NOP/NOT Program. The assumptions delineated in subsection 5.5.2.2 have been included in the containment model such that conservative containment pressures and lower compartment temperatures are calculated. The containment pressure and temperature acceptance criteria have been met for all MSLB cases analyzed for the Cook Unit 1 Return to RCS NOP/NOT Program.

5.5.2.6 References

1. AEP Letter AEP:NRC:2900, "Donald C. Cook Nuclear Plant Unit 1 Docket No. 50-315, License Amendment Request for Appendix K Measurement Uncertainty Recapture – Power Uprate Request," June 28, 2002.

2. USNRC Letter, "Donald C. Cook Nuclear Plant, Unit 1 - Issuance of Amendment 273 Regarding Measurement Uncertainty Recapture Power Uprate (TAC No. MB5498)," December 20, 2002.

3. WCAP-8354-P-A (Proprietary) and WCAP-8355-A (Nonproprietary), "Long Term Ice Condenser Containment Code-LOTIC Code," April 1976.

4. WCAP-8354-P-A, Supplement 1 (Proprietary), and WCAP-8355-A, Supplement 1 (Nonproprietary), "Long Term Ice Condenser Containment Code - LOTIC Code," April 1976.

5. WCAP-8354-P-A, Supplement 2 (Proprietary), and WCAP-8355-NP, Supplement 2 (Nonproprietary), "Westinghouse Long Term Ice Condenser Containment Code – LOTIC-3 Code," February 1979.

Table 5.5.2-1 Cook Unit 1 Containment Response Results for the MSLB Event

Power Level	Break Type	Break Size	MSLB Single Failure	Peak Pressure @ Time	Peak Temperature @ Time
100.34%	DER	1.4 ft^2	MSIV	9.61 psig @ 12.76 sec	323.80°F @ 55.57 sec
100.34%	DER	1.4 ft^2	AFW Runout Control	9.61 psig @ 12.76 sec	323.71°F @ 54.76 sec
70%	DER	1.4 ft^2	MSIV	9.60 psig @ 12.16 sec	323.78°F @ 12.36 sec
70%	DER	1.4 ft^2	AFW Runout Control	9.60 psig @ 12.16 sec	323.75°F @ 12.36 sec
30%	DER	1.4 ft^2	MSIV	9.72 psig @ 2.46 sec	324.32°F @ 12.36 sec
30%	DER	1.4 ft^2	AFW Runout Control	9.72 psig @ 2.46 sec	324.35°F @ 12.36 sec
0%	DER(1)	1.4 ft^2	MSIV	9.72 psig @ 2.46 sec	320.89°F @ 131.3 sec
0%	DER(1)	1.4 ft^2	AFW Runout Control	9.72 psig @ 2.46 sec	320.35°F @ 132.8 sec
0%	DER	1.0 ft^2	MSIV	8.31 psig @ 20.22 sec	324.49°F @ 265.4 sec
0%	DER	1.0 ft^2	AFW Runout Control	7.86 psig @ 13.81 sec	324.11°F @ 140.8 sec
100.34%	Split	0.865 ft^2	MSIV	6.77 psig @ 56.96 sec	324.67°F @ 96.57 sec
100.34%	Split	0.865 ft^2	AFW Runout Control	6.75 psig @ 48.91 sec	324.49°F @ 95.86 sec
70%	Split	0.857 ft^2	MSIV	6.93 psig @ 56.58 sec	324.66°F @ 72.43 sec
70%	Split	0.857 ft^2	AFW Runout Control	6.82 psig @ 53.23 sec	324.50°F @ 111.4 sec
30%	Split	0.834 ft^2	MSIV	6.92 psig @ 63.93 sec	324.60°F @ 122.9 sec
30%	Split	0.834 ft^2	AFW Runout Control	6.78 psig @ 57.90 sec	324.29°F @ 116.7 sec
0%	Split	0.808 ft^2	MSIV	6.73 psig @ 62.55 sec	324.29°F @ 142.8 sec
0%	Split	0.808 ft^2	AFW Runout Control	6.68 psig @ 58.29 sec	324.54°F @ 119.7 sec

Note:
1. All MSLB M&E releases assume dry saturated steam except this break, which assumes entrainment liquid in the M&E releases from the faulted SG

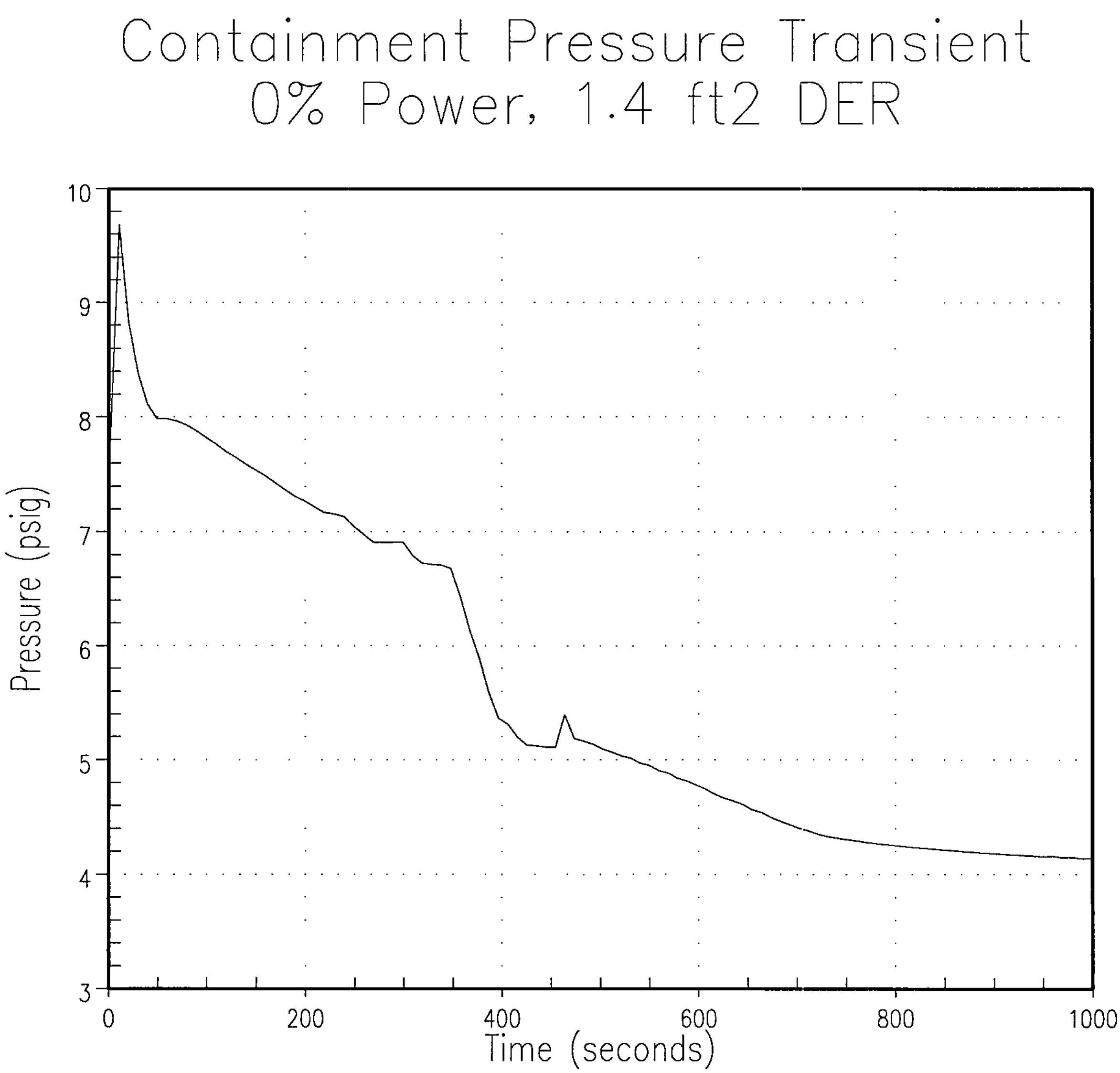


Figure 5.5.2-1 Steamline Break Containment Pressure Response to a 0% Power, 1.4 ft^2 DER

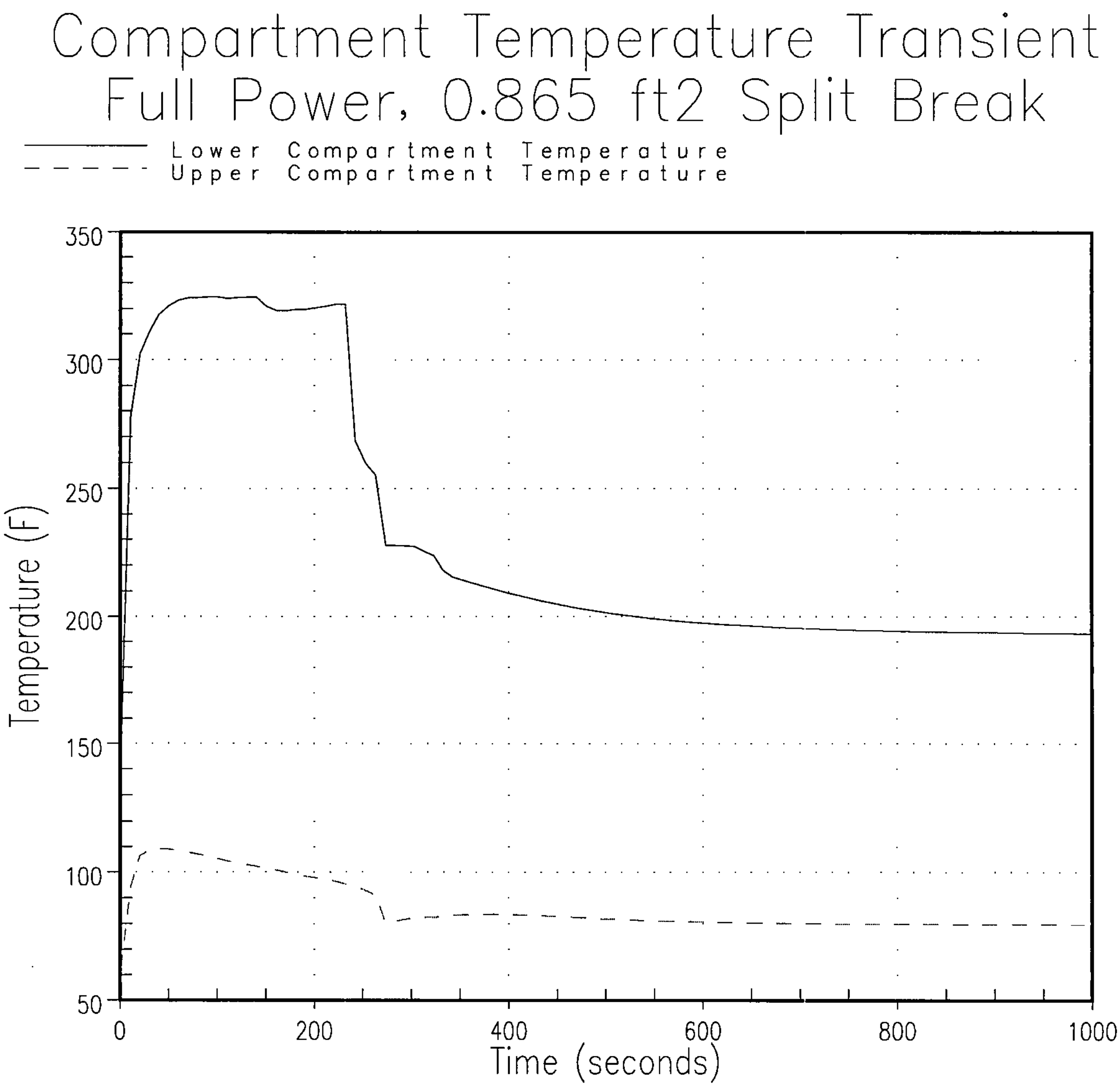


Figure 5.5.2-2 **Steamline Break Compartment Temperature Responses to a Full Power (100.34%), 0.865 ft^2 Split Break with an MSIV Failure**

6 NUCLEAR FUEL

6.1 NUCLEAR DESIGN

Cycle-specific calculations are performed for each reload cycle. These cycle-specific analyses and evaluations are performed to demonstrate that all core design and Reload Safety Analysis Checklist (RSAC) criteria will be satisfied for the specific operating conditions of that cycle.

The standard set of reload core design criteria (Reference 1) has been confirmed via evaluation or explicit analysis for the transition to Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) conditions. For RSAC items analyzed for each cycle, adequate margins to the limits have been demonstrated for recent cycles to provide assurance that these limits will not be challenged by the transition to NOP/NOT conditions or, if needed, they were explicitly analyzed.

For the hot full power steamline break (a new RSAC item for Cook Unit 1), doppler power coefficient, and burnup dependent peaking factors explicit calculations were performed, using input values from three cycles' worth of ANC scoping models that were at NOP/NOT conditions. These calculations incorporated updated Non-LOCA and LOCA inputs, which accounted for NOP/NOT conditions where necessary. The impact of NOP/NOT conditions on all RSAC parameters that were not explicitly analyzed was evaluated against typical margins to their limits. Adequate margin was confirmed to be available.

6.1.1 References

1. Westinghouse Report WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

6.2 FUEL ROD DESIGN

6.2.1 Introduction and Background

The fuel performance for Cook Unit 1 at normal operating pressure/normal operating temperature (NOP/NOT) conditions shall satisfy the United States Nuclear Regulatory Commission (USNRC) fuel rod design bases on a region-by-region basis. These same bases are applicable to all fuel rod designs, including Westinghouse 15x15 Optimized Fuel Assembly (OFA). This licensing basis analysis is based on maintaining the current fuel, 15x15 OFA with ZIRLO® and **Optimized ZIRLO™** High Performance Fuel Cladding Material[1], and the nuclear design information described in Section 6.1, Nuclear Design.

The current licensing basis is described in Chapter 3, Unit 1 Reactor, of the D. C. Cook Updated Final Safety Analysis Report (UFSAR). Compliance with USNRC General Design Criteria (GDC) 10 specified acceptable fuel design limits (SAFDLs) for reload cycles is confirmed via the approved Westinghouse reload methodology (Reference 1).

1. ZIRLO® and **Optimized ZIRLO™** are trademarks or registered trademarks of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States of America and may be registered in other countries throughout the world. All rights reserved. Unauthorized use is strictly prohibited. Other names may be trademarks of their respective owners.

6.2.2 Input Parameters and Assumptions

Fuel rod design evaluations are performed using USNRC-approved models (References 3 and 5) and USNRC-approved design criteria methods (References 2, 4, 6, 7, and 8) to demonstrate that all fuel rod design criteria are satisfied. The fuel rod design criteria given below are verified by evaluating the predicted performance of the limiting fuel rod, defined as the rod that has the minimum margin to the design limit, on a reload-specific basis. In general, no single rod is limiting with respect to all of the design criteria.

The current fuel performance and design models (PAD 4.0) are also used to generate fuel temperature and rod internal pressure (RIP) data for loss-of-coolant accident (LOCA) safety evaluations.

6.2.3 Acceptance Criteria

The conditions and requirements related to fuel rod design analyses are summarized below:

1. The fuel rod burnup limit remains at 62 GWD/MTU.

2. The maximum fuel rod waterside corrosion and the calculated metal-oxide interface temperatures will be less than the licensed limits (Reference 7).

3. All the conditions listed in previous USNRC safety evaluation approvals for methodologies used for standard ZIRLO® and Zircaloy-4 cladding fuel analysis will continue to be met (Reference 7).

4. The relative differences in unirradiated strength (yield and ultimate) between **Optimized ZIRLO™** and standard ZIRLO® cladding and structural analyses will be accounted for until irradiated data for **Optimized ZIRLO™** have been obtained and provided to the USNRC.

The fuel rod performance evaluations for Cook Unit 1 at NOP/NOT conditions shall account for the reduction in unirradiated strength between standard ZIRLO® and **Optimized ZIRLO™** cladding.

The criteria pertinent to the fuel rod design are described below:

6.2.3.1 Rod Internal Pressure (Gap Reopening and DNB Propagation)

The internal pressure of the lead rod in the reactor will be limited to a value below that which could cause the diametral gap to increase due to outward cladding creep during steady-state operation or for extensive departure from nucleate boiling (DNB) propagation to occur.

6.2.3.2 Cladding Stress and Strain

The volume average effective clad stress with the Von Mises equation considering interference due to uniform cylindrical pellet-clad contact, caused by pellet thermal expansion, pellet swelling and uniform clad creep, and pressure differences is less than the 0.2 percent offset yield stress with due consideration to temperature and irradiation effects under Condition I and II events.

The design limit for cladding strain during steady-state operation is that the total plastic tensile creep strain due to uniform cladding creep and uniform cylindrical fuel pellet expansion associated with fuel swelling and thermal expansion is less than 1percent from the unirradiated condition. The design limit for fuel rod cladding strain during Condition II events is that the total tensile strain due to uniform cylindrical pellet thermal expansion is less than 1percent from the pre-transient value.

6.2.3.3 Cladding Oxidation and Hydriding

The design criteria related to cladding corrosion require that the ZIRLO® and **Optimized ZIRLO™** cladding metal-oxide interface temperature, oxide thickness, and hydrogen pickup be maintained below specified limits to prevent a condition of accelerated oxidation, which would lead to cladding failure.

6.2.3.4 Fuel Temperature

The design limit for fuel temperature analyses during Condition I and Condition II events is that the fuel centerline temperature will not exceed the fuel melting temperature criterion. The intent of this criterion is to avoid a condition of gross fuel melting, which can result in severe duty on the cladding. The concern here is based on the large volume increase associated with the phase change in the fuel and the potential for loss of cladding integrity as a result of interaction of the molten fuel with the cladding.

6.2.3.5 Cladding Fatigue

The design limit for cladding fatigue is that for a given strain range, the number of strain fatigue cycles are less than those required for failure, considering a minimum safety factor of 2 on the stress amplitude and a minimum safety factor of 20 on the number of cycles. The fatigue life usage factor shall be less than 1.0.

The concern of this criterion is the accumulated effect of short-term, cyclic, cladding stress and strain which result primarily from daily load follow operation.

6.2.3.6 Cladding Flattening

The design limit is that the fuel rod shall preclude cladding flattening during the projected exposure. This criterion was established to prevent the long-term creep collapse of the fuel rod into axial gaps that can form within the fuel column. Current fuel rod designs employing fuel with improved in-pile stability provides adequate assurance that axial gaps large enough to allow cladding flattening will not form within the fuel stack.

6.2.3.7 Fuel Rod Axial Growth

The design limit is that the space between the rod end plug-to-end plug outer dimension and the lower nozzle-to-top adaptor plate inner dimension shall be sufficient to preclude interference of these members. The evaluation considers the effects of fuel rod growth, thimble growth and creep, and thermal expansion of these members. Contact is to be precluded to avoid overstressing of thimble tubes and/or thimble-to-nozzle joints.

6.2.3.8 Plenum Cladding Support

The design limit ensures that the fuel rod cladding in the plenum region will not collapse during the lifetime of the fuel under normal operating conditions.

6.2.3.9 Cladding Free Standing

The design limit requires that the cladding be short-term free standing at beginning of life (BOL), at power, and during hot hydrostatic testing. This criterion precludes the instantaneous collapse of the cladding onto the fuel pellet caused by the pressure differential that exists across the cladding wall.

6.2.3.10 Fuel Rod End Plug Weld Integrity

The fuel rod end plug weld shall maintain its integrity during Condition I and II events and shall not contribute to any additional fuel failures above those already considered for Condition III and IV events. The intent of this criterion is to assure that fuel rod failures will not occur due to the tensile pressure differential loads that can exist across the weld.

6.2.4 Description of Analyses and Evaluations

The PAD code with USNRC-approved models (References 3 and 5) for in-reactor behavior is used to calculate the fuel rod performance over its irradiation history. PAD is the principal design tool for evaluating fuel rod performance. PAD iteratively calculates the interrelated effects of temperature, pressure, cladding elastic and plastic behavior, fission gas release, and fuel densification and swelling as a function of time and linear power.

PAD 4.0 is a best-estimate fuel rod performance model, and in most cases the design criterion evaluations are based on a best-estimate plus uncertainties approach. A statistical convolution of individual uncertainties due to design model uncertainties and fabrication dimensional tolerances is used. As-built dimensional uncertainties for some critical inputs, such as fuel pellet diameter, can be used in lieu of the fabrication uncertainties.

No explicit PAD calculations were performed to evaluate the fuel rod design criteria at NOP/NOT conditions. All fuel rod design criteria described in subsection 6.2.3, Acceptance Criteria, shall be evaluated for Cook Unit 1 at NOP/NOT conditions on a cycle-specific basis to ensure that they are met.

6.2.4.1 Rod Internal Pressure (Gap Reopening and DNB Propagation)

Increasing the vessel average temperature as part of the Return to RCS NOP/NOT Program can cause higher fuel temperatures via a reduction in heat transfer capability between the fuel rod and the coolant. However, the increased system pressure causes faster pellet-clad contact, increasing the thermal conductivity of the fuel, and thereby reducing additional fission gas release. Any increase in fuel temperature or rod internal pressure resulting from the competing effects of implementing the Return to RCS NOP/NOT Program are therefore considered to be marginal and can be offset by existing rod internal pressure (RIP) margin.

6.2.4.2 Cladding Stress and Strain

Increased coolant temperature and system pressure resulting from the NOP/NOT conditions negatively impacts the cladding stress and strain margins. However, use of constant axial offset control (CAOC) bands restricts the Condition I and II transient events that are limiting for cladding stress and strain. The continued use of CAOC bands, along with the available margin to the cladding stress and strain limits, is sufficient to confirm that the design criteria can remain met at NOP/NOT conditions.

6.2.4.3 Cladding Oxidation and Hydriding

As discussed in subsection 6.2.4.1, the Return to RCS NOP/NOT Program has competing effects on the fuel and cladding temperature. As these effects are considered minor, and corrosion is a direct function of cladding temperature, the impacts of the Return to RCS NOP/NOT Program can be offset by available margin.

6.2.4.4 Fuel Temperature

The behavior of a fuel rod as a function of burnup is simulated in PAD by depleting the fuel rod using an input power history that approximates the environment seen by the fuel over its lifetime. At various times during the analysis, the depletion is stopped and the power is rapidly ramped down to near-zero power and then ramped up to a specified peak linear power. In this manner, fuel temperatures and RIPs as a function of both linear power and burnup are determined. Fuel temperature and RIP analyses used as input to LOCA and transient safety evaluations are not loading pattern dependent and remain valid for any core loading strategy implemented at NOP/NOT conditions. The core stored energy has also been determined for use in containment analysis. Core stored energy is defined as the amount of energy in the fuel rods in the core above the local coolant temperatures. The local core stored energy is normalized to the local linear power level. The applicability of these values is confirmed on a cycle-specific basis.

6.2.4.5 Cladding Fatigue

As with cladding stress and strain, the effects of implementing the Return to RCS NOP/NOT Program will not have a significant impact on cladding fatigue. There is sufficient available margin to the design limit to offset any slight increase in cladding fatigue.

6.2.4.6 Cladding Flattening

Cladding flattening is caused by axial gaps that develop in the fuel column. Westinghouse fuel has been shown to not form axial gaps in the fuel stack, and therefore there is no impact to this design criterion as a result of the Return to RCS NOP/NOT Program.

6.2.4.7 Fuel Rod Axial Growth

Fuel rod axial growth is calculated as a function of fluence and burnup, which are not affected by operating at NOP/NOT conditions. Therefore, there is no impact to fuel rod axial growth design criterion.

6.2.4.8 Plenum Cladding Support

The plenum spring has been shown to provide sufficient support for the cladding in the plenum region. As the spring dimensions do not change, the plenum cladding support is not impacted by the change to NOP/NOT conditions.

6.2.4.9 Cladding Free Standing

The critical collapse pressures are in excess of the maximum expected differential pressure across the clad for all Westinghouse fuel rod geometries. Therefore, the cladding free standing design criterion is not impacted by operation at NOP/NOT conditions.

6.2.4.10 Fuel Rod End Plug Weld Integrity

As discussed in subsection 6.2.4.1, there is marginal impact to RIP as a result of NOP/NOT operation. End plug weld integrity is limited by higher pressure differentials during depressurization events. Marginal changes to the rod internal pressure as a result of operating at NOP/NOT conditions can be offset by the available margin.

6.2.5 Results

The PAD code with USNRC-approved models was used to generate fuel temperature and RIP data used as input to LOCA and transient safety evaluations. It was also used to calculate core stored energy used as input to containment analyses.

The remaining fuel rod design criteria are evaluated for Cook Unit 1 at NOP/NOT conditions on a reload-specific basis to ensure that all of the design limits are met. The impacts of operating at NOP/NOT conditions either have no impact on the fuel rod design criteria or can be offset by available margin.

6.2.6 Conclusions

All fuel rod design criteria will continue to be met for Cook Unit 1 at NOP/NOT operating conditions using USNRC-approved models and methods, per the Westinghouse reload methodology established in Reference 1, and the discussion of each design criterion provided in subsection 6.2.4.

6.2.7 References

1. Westinghouse Report WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

2. Westinghouse Report WCAP-10126-NP-A, "Extended Burnup Evaluation of Westinghouse Fuel," December 1985.

3. Westinghouse Report WCAP-11873-A, "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," August 1988.

4. Westinghouse Report WCAP-14297-A, "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel," March 1995.

5. Westinghouse Report WCAP-15064-NP-A, Revision 1 with Errata, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," July 2000.

6. Westinghouse Report WCAP-14342-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995.

7. Westinghouse Report WCAP-14204-A, "Westinghouse Fuel Criteria Evaluation Process," October, 1994 and WCAP-14204-A, Addendum 1-A, Revision 1, "Addendum 1 to WCAP-14204-A Revision to Design Criteria," January 2002.

8. Westinghouse Report WCAP-14342-A & CENPD-404-NP-A, Addendum 1-A, "**Optimized ZIRLO™**," July 2006.

6.3 CORE THERMAL & HYDRAULIC DESIGN

6.3.1 Introduction and Background

This section describes the core thermal-hydraulic (T/H) analysis performed to support the Cook Unit 1 Return to Reactor Coolant System (RCS) Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) Program. In support of this program, the current T/H design licensing basis events were reviewed and reanalyzed when the current licensing basis did not conservatively bound the NOP/NOT operational conditions. The only event that took credit for the current operating pressure and temperature was the hot full power (HFP) steamline break (SLB) event, which was analyzed explicitly as part of the Return to RCS NOP/NOT Program. All other events described in the current Cook Unit 1 licensing basis conservatively bound the Return to RCS NOP/NOT Program operating conditions.

The current design methodology for Cook Unit 1 Reload Safety Evaluations (RSE) in Reference 4 remains unchanged for the NOP/NOT evaluation. The USNRC-approved WRB-1 departure from nucleate boiling (DNB) correlation continues to be used for DNB analysis while the W-3 DNB correlation is used where the conditions fall outside the applicable range of the WRB-1 correlation.

6.3.2 Input Parameters and Assumptions

Analyses are conducted using nominal values consistent with full power operation and the minimum measured flow value. Reactor power, pressure, and inlet temperature values for the HFP SLB event are used as determined in Section 5.2, Non-LOCA Transients.

6.3.3 Hot Full-Power Steam Line Break Accident

The DNB analysis was conducted for the HFP SLB according to the USNRC-approved methodology described in Reference 1. Departure from nucleate boiling ratio (DNBR) calculations for the HFP SLB accident at NOP/NOT conditions were performed using the Westinghouse version of the VIPRE-01 code, VIPRE-W (Reference 2), the WRB-1 correlation, and the Revised Thermal Design Procedure (RTDP) methodology (Reference 3). VIPRE-W is the configured quality assurance (QA) version of the Westinghouse VIPRE-01 code and VIPRE-W has been evaluated to be in full compliance with the VIPRE-01 methodology in Reference 2, including the results and conclusions stated and approved in Reference 2. Acceptance criteria for the HFP SLB event are described in Section 5.2.

6.3.3.1 HFP SLB Results and Conclusions

The results of the DNB analysis of the limiting statepoint for the HFP SLB transient confirmed that the calculated minimum DNBR remains greater than the RTDP design limit DNBR of 1.21, which is based on the WRB-1 correlation DNBR Limit of 1.17 from Reference 2. Therefore, the DNB design basis continues to be met with the implementation of the Return to RCS NOP/NOT Program. Confirmation of the DNB design basis demonstrates that Safety Evaluation Report (SER) acceptance criteria specified in Reference 2, which are already included in the Cook Unit 1 licensing basis, continue to be met.

6.3.4 References

1. Westinghouse Report WCAP-9226-P-A Rev. 1, "Reactor Core Response to Excessive Secondary Steam Releases," February 1998.

2. Westinghouse Report WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

3. Westinghouse Report WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

4. Westinghouse Report WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

6.4 FUEL MECHANICAL DESIGN

In support of the Cook Unit 1 Return to Reactor Coolant System (RCS) Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) conditions of 2250 psia and T_{avg} of 571°F, respectively, the fuel mechanical design analyses were reviewed to determine if the return to NOP/NOT can be supported.

The fuel mechanical design analyses, specifically the Fuel Assembly Grid Load Analysis, are dependent on parameters such as the fuel assembly (FA) burnup, and make bounding assumptions for the nuclear steam suppy system (NSSS) operating conditions. As a result, they are not affected by Cook Unit 1's Return to RCS NOP/NOT Program. The only exception is the FA lift force calculation which is used as input to the top nozzle holddown spring force evaluation.

The FA lift force calculation was specifically performed for the conditions associated with the return to NOP/NOT. The FA lift force calculations use a bounding (high) best estimate flow rate which maximizes the FA lift forces, as well as a bounding (low) RCS temperature as this maximizes the density and thus the FA lift forces. In addition, a minimum bypass flow with thimble plugs inserted is assumed as this also maximizes the FA flow rate and thus the FA lift forces. The fuel assembly lift forces are used in the top nozzle holddown spring force evaluation. The top nozzle holddown spring force evaluation demonstrates that FA liftoff will not occur except during a pump overspeed condition. An examination of the FA lift forces assumed in the top nozzle holddown spring force analysis indicates that the current analysis of record FA lift forces bound the FA lift forces associated with the Cook Unit 1 Return to RCS NOP/NOT Program.

Therefore, the existing/current fuel mechanical design analyses, and specifically the top nozzle holddown spring force analysis remain valid and bounding for the Cook Unit 1 Return to RCS NOP/NOT Program.

Enclosure 7 to AEP-NRC-2013-79

Programs and Systems Evaluations

Programs and Systems Evaluations

Evaluations of Cook Plant programs and non-Reactor Coolant System (RCS) systems were conducted to determine the impacts from operating Unit 1 at the proposed RCS operating pressure of 2250 psia and average full power temperature (T_{avg}) of 571°F, referred to as Return to Normal Operating Pressure and Temperature (NOP/NOT). It is particularly noteworthy that, where applicable, the Unit 1 systems, structures, and components were originally designed to accommodate an RCS normal operating pressure and average temperature similar to the new values proposed in this License Amendment Request. Summaries of the completed evaluations are provided in this Enclosure.

High Energy Line Break (HELB)

Per the HELB analysis of record for Unit 1, HELB mass and energy releases are based on a Nuclear Steam Supply System (NSSS) power of 3426 MWt and a steam pressure of 820 psia, which bounds the proposed changes. Therefore, no changes to HELB zones or barriers are required. The HELB program is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Flow Accelerated Corrosion (FAC)

A revised secondary side heat balance showing operation at NOP/NOT conditions will be provided to the FAC program coordinator prior to closeout of the Engineering Change package that implements the proposed change in RCS operating conditions. This heat balance will serve as input to the FAC model, and appropriate changes to the FAC program will be incorporated as necessary. Therefore, the FAC program is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Spent Fuel Pool (SFP) Cooling System

The primary function of the SFP cooling system is to remove decay heat that is generated by the spent fuel assemblies stored in the pool. Decay heat generation is proportional to plant power level. Since the reactor thermal power level of 3304 MWt remains unchanged, the demands on the SFP cooling system are not increased. The purification function is controlled by SFP cooling system demineralization and filtration rates that are not affected by the proposed increase in RCS pressure and average temperature. It was concluded, therefore, that the SFP cooling system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Makeup Water System

The condensate makeup water system is designed for continuous service and is a shared system supplying demineralized water to both Unit 1 and Unit 2. The makeup water system contains three demineralizer trains, each containing a cation exchanger, an anion exchanger, and a mixed bed demineralizer. There is a vacuum degasifier common to all demineralizer trains. Each train is designed for a maximum flow of 400 gpm and a normal flow of 300 gpm. The makeup water system demand is not affected by the proposed increase in RCS pressure and average temperature. The makeup water system is adequately sized to handle small changes in normal flash tank blowdown rates. It was concluded, therefore, that the makeup water system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Turbine Room Drainage System (Secondary Waste Liquid)

The purpose of the turbine room drainage system is to collect, neutralize and dispose of non-radioactive liquids from the turbine building through floor drains, equipment drains, and roof drains. The turbine room drainage system consists of sumps and the associated piping and pumps. The increased primary temperature and pressure will not increase the volume of flow or flow rate of non-radioactive liquids in the turbine room drainage system. It was concluded, therefore, that the turbine room drainage system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Nuclear Sampling System

The nuclear sampling system is designed to provide representative samples from the pressurizer, hot legs, and blowdown lines for laboratory analyses. Sample results are used to guide the operation of various primary and secondary systems throughout the plant during normal operation. The heat exchangers and needle valves used to reduce sample pressure and temperature were designed for process conditions that encompass the proposed NOP/NOT conditions. It was concluded, therefore, that the nuclear sampling system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Steam Generator Blowdown and Blowdown Treatment Systems

The steam generator blowdown system is used mainly to control secondary side water chemistry. It is also used to drain the steam generators during plant outages. The steam generator blowdown treatment system is used in the event of a primary to secondary steam generator tube leak to remove radioactive ions and particulates. This provides for continued blowdown usage while preventing radioactive release.

Currently, the blowdown rate is controlled by the operator depending on system conditions. At an increased primary temperature and pressure, the blowdown system will continue to perform its function as required to maintain the proper secondary side water chemistry. The treatment system will continue to operate based on the original design when the plant is restored to normal operating temperature and pressure.

It was concluded, therefore, that the steam generator blowdown system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Chemical Feed System

The condensate and feedwater chemical feed systems supply the appropriate amount of chemical additive to the condensate and feedwater. The increase of primary temperature and pressure will not significantly change the feedwater flow rate; therefore, it was concluded that the condensate and feedwater chemical feed systems are not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Auxiliary Feedwater (AFW) System

The AFW system provides water to the steam generators when the main feedwater system is not available due to a loss of main feedwater, unit trip, feedwater or steam line break, loss of off-site power, or loss of coolant accident (LOCA). The source of water is the condensate storage tank (CST) or the Essential Service Water System (emergency water source) if the CST is unavailable. The AFW system also provides water during start-up and shutdown when sufficient steam is not available to drive the main feed pumps.

The AFW system is designed, in conjunction with the CST, to remove residual heat from the reactor core upon loss of main feedwater. The system functions to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. The AFW system consists of one turbine-driven auxiliary feed pump, which feeds all four steam generators, and two motor-driven auxiliary feed pumps, each of which feeds two steam generators. Design basis flow rates for the AFW system pumps were established prior to original plant licensing when the RCS normal operating pressure and the full power T_{avg} range were comparable to the proposed request in this License Amendment Request.

In addition, the accident analyses and evaluations performed to support this License Amendment Request (see Enclosure 6 of this letter) used the existing AFW system performance capabilities and satisfied all analyses acceptance criteria.

It was concluded, therefore, that the AFW system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Containment Spray (CTS) System

The CTS system provides spray cooling water to the containment atmosphere during a LOCA or steam line break accident. This cooling water limits the peak pressure in the containment to below the containment design pressure (12 psig) and maintains the peak temperature within Environmental Qualification limits. A secondary function of the spray system is to remove radioactive iodine that would be released into containment during a break of the fuel cladding following a LOCA.

In addition to the proposed increase in RCS normal operating pressure and full power T_{avg}, the setpoint of the time delay relay in each of the two CTS pump starting circuits is being revised to delay CTS system actuation following a LOCA or steam line break. This change helps offset the adverse impact of the proposed increase in full power T_{avg} on the Best Estimate (BE) LOCA peak cladding temperature (PCT) with thermal conductivity degradation effects accident analysis.

The effects of the proposed increase in RCS normal operating pressure and full power T_{avg} on the Unit 1 accident analyses were addressed by Westinghouse and are described in Sections 5.4 and 5.5 of WCAP-17762-NP, Rev. 1 (Enclosure 6 of this letter). Impacts on the control room habitability and offsite dose consequence analyses were addressed by I&M and are documented in Enclosure 8 of this letter.

As a result of the reviews described in Enclosures 6 and 8 of this letter, it was concluded that the containment spray system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}, and the proposed changes in the setting of the CTS pump time delay relays ensure that the acceptance criteria of the accident and radiological consequence analyses continue to be met.

Component Cooling Water (CCW) System

The CCW system is a safety related, closed loop cooling system which serves as an intermediate system between potentially radioactive heat sources and the ultimate heat sink (Lake Michigan) to ensure that leakage of radioactive fluid from components being cooled is contained within the plant. The CCW System is designed to: a) remove residual and sensible heat from the RCS via the Residual Heat Removal (RHR) system during shutdown; b) cool the spent fuel pool water and the letdown flow to the Chemical Volume and Control System (CVCS) during power operation; c) dissipate waste heat from various primary plant components; and d) provide cooling for safeguards equipment.

The proposed increase in normal RCS operating pressure and full power T_{avg} will have little, if any, effect on CCW heat loads during normal operation. To address the proposed full power T_{avg} increase to 571°F, the pressurizer water level control program will be revised, consistent with existing Westinghouse design basis, to satisfy the program's primary goal of maintaining a relatively constant CVCS charging rate. As a result, normal CVCS letdown flow rates are not expected to change due to NOP/NOT, thereby maintaining similar CCW heat loads from the letdown heat exchanger.

In addition, since reactor thermal power is not changing, the decay heat loads, which must be removed during normal cooldown operations and in response to postulated accident mitigation, will not change. Finally, as noted in UFSAR Table 9.5-2, Component Cooling Water System Flow Requirements per Train (GPM), the CCW system design flow requirements are highest during LOCA recirculation and RCS cooldown below 350°F and not during normal operating conditions. Plant cooldown and accident-related CCW heat loads are not adversely affected by the changes proposed in this License Amendment Request.

As a result, it was concluded that the CCW system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Essential Service Water (ESW) System

The ESW system provides the cooling water requirements for the component cooling water heat exchangers, the emergency diesel generator coolers, the containment spray heat exchangers, the control room air conditioning condensers, and the auxiliary feedwater pump enclosure coolers. The ESW system, shared by Units 1 and 2, consists of four ESW pumps, each with an automatic backwashing duplex strainer and associated piping, valves, and instrumentation. The ESW system is comprised of two identical main headers. Each header is served by two pumps and each header, in turn, serves half of the system load in each unit.

Cooling requirements for the emergency diesel generator coolers, control room air conditioner condensers, and auxiliary feedwater pump enclosures are not a function of RCS conditions, and are thus unaffected by the proposed increase in RCS normal operating pressure and full power T_{avg}. Since reactor thermal power is not changing, the cooling requirements for the component cooling water and containment spray heat exchangers are not affected by the proposed RCS conditions. As a result, it was concluded that the ESW system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Nonessential Service Water (NESW) System

The NESW system is a shared system for Units 1 and 2 that provides Lake Michigan water as cooling and makeup water to numerous plant systems and components.

The system consists of four NESW supply pumps (two per unit) each with an automatic backwashing duplex strainer, cooling water suction supply lines from the circulating water intake and discharge tunnels, cooling water lines to the various components being serviced, and associated valves and instrumentation. NESW flows from the pumps to the equipment served and is then returned to Lake Michigan via the circulating water discharge tunnel. Since reactor thermal power is not changing, the heat loads from the proposed NOP/NOT conditions do not differ significantly from the present conditions. Therefore, the heat loads to the NESW system are not significantly affected. As a result, it was concluded that the NESW system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Containment Chilled Water Subsystem (subsystem of NESW)

The Containment Chilled Water Subsystem, added as an enhancement to plant operation in 2011, consists of a closed-loop chilled water system and an open-loop condenser cooling system designed to augment containment cooling. Since the RCS average temperature will increase under NOP/NOT conditions, slightly more heat will be lost from the RCS to the containment atmosphere, which will place a higher heat load on the chillers during operation. However, the heat loads used as design inputs for this subsystem were based on historical data for both Unit 1 and Unit 2, with bounding values selected for a common design. Because Unit 2 has always operated at RCS conditions representative of the proposed Unit 1 NOP/NOT conditions, the design of the subsystem remains acceptable for the proposed Unit 1 NOP/NOT conditions. As a result, it was concluded that the Containment Chilled Water Subsystem is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Feedwater Heater Extractions, Drains, and Vents

Drains from low pressure feedwater heater Nos. 1, 2, 3, and 4 cascade to the drain cooler, which drains to the main condenser. No. 6 feedwater heater drains cascade to the shell of the No. 5 heater, which drains to the heater drain pumps. Two of three 50 percent capacity pumps inject this drain flow into the suction of the main feed pumps. Level control valves on the low pressure heater drain lines and on high pressure heater No. 6 automatically maintain the normal water level in the heaters. High pressure heater No. 5 level is maintained by the heater drain pump discharge control valves.

Under NOP/NOT conditions the drain flows from heaters 1, 2, 3, and 4 will be slightly decreased with no appreciable change in the amount of non-condensable gases. As a result of the small decrease in drain flow, the level control valves will modulate further closed to maintain the desired water level in the heaters. Setpoint changes for the level controllers will not be required. As the drain flow from heaters 5 and 6 will be slightly increased, the heater drain pump will maintain sufficient capacity. The vent system that removes the non-condensable gases from the shell side of the heaters will not be affected. Therefore, neither the drains nor venting system will require modification under the proposed operating conditions.

As a result, it was concluded that the feedwater heaters extractions, drains, and vents are not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Main Feedwater System

The main feedwater system consists of two feed pump suction strainers, two main feed pumps, two parallel strings of No. 5 and 6 high pressure heaters, four feedwater control valves and associated piping, valves and instrumentation. The feedwater pumps take suction through the strainers from a common header supplied by the outlet flow from the low pressure heaters and the discharge from the heater drain pumps. Prior to being pumped to the steam generators, the feedwater is passed through the high pressure heaters where additional heat is added to the system.

The slight changes in flow, pressure, and temperature associated with the proposed operating conditions remain within the design limits of the system components. As a result, it was concluded that the main feedwater system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Condensate System (CS)

The CS, in conjunction with the feedwater system, returns the condensed steam from the condensers and the feedwater heater drains to the steam generators while maintaining the overall water inventory throughout the cycle. The system is also required to compensate for the loss of fluid from the steam cycle when an atmospheric steam dump occurs. The necessary water inventory is maintained in the condenser hotwells and the condensate storage tank.

The condensate system for Unit 1 consists of one main condenser (separated into three shells, one for each low pressure turbine), three hotwell pumps, three condensate booster pumps, four steam jet air ejectors, the turbine auxiliary cooling cycle, four stages of low pressure feedwater heating, the condensate storage tank and associated instrumentation, piping, and valves.

The thermal load on the condenser remains essentially the same under the proposed operating conditions, as will condensate temperature. As a result, it was concluded that the CS is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Extraction Steam System

The extraction steam system (bleed steam system) provides a source of steam to heat the condensate and feedwater, and also supplies steam to the auxiliary steam system.

At NOP/NOT conditions, the bleed steam system will experience lower velocities due to the higher steam pressures. Operation will remain within the original design parameters for the system. As a result, it was concluded that the extraction steam system will not be adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Circulating Water (CW) System

The CW system is an open loop system that provides a heat sink for waste heat from the plant thermal cycle. The CW system supplies cooling water to various coolers and condensers during all phases of plant operation.

Lake Michigan water is piped into the forebay from which the CW pumps take suction. The pumps circulate the CW through the various services before it is returned to the lake.

With a slightly lower steam flow into the main condenser as a result of the increased steam pressure, the temperature rise across the condenser will be decreased. However, the decrease is so slight that the CW system will not be affected. As a result, it was concluded that the circulating water system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Main Condenser Evacuation System

The steam jet air ejector system provides for air evacuation and removal of non-condensable gases from the main and feed pump turbine condensers to promote high condenser efficiency.

The steam jet air ejector system for each unit consists of three single element, single stage startup ejectors and four twin element, two-stage holding ejectors. The startup ejectors pull the vacuum in the condensers to approximately 15 inches Hg. From this point, the holding ejectors pull the condenser vacuum down to operating vacuum and maintain it at this level. With condenser conditions remaining essentially the same under NOP/NOT conditions, the current evacuation capacity, which includes 100 percent spare ejector capacity for the ejectors serving the main condensers, is more than adequate. As a result, it was concluded that the steam jet air ejector system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Turbine Auxiliary Cooling Water (TACW) System

The TACW system is a subsystem of the condensate system. It uses main cycle condensate to remove heat from various heat exchangers associated with the turbine-generator unit. The TACW system consists of two turbine auxiliary cooling pumps, one turbine auxiliary cooler and various other heat exchangers in the turbine-generator unit, which are provided cooling water by the system.

Condensate temperature and supply flow rates to the TACW system from the condensate system will not change and heat loads from the turbine-generator unit heat exchangers will not increase under NOP/NOT conditions. As a result, it was concluded that the TACW system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Main Turbine and Feed Pump Turbines (FPT)

The main turbine consists of one double-flow high pressure (HP) turbine and three double-flow low pressure (LP) turbines. Increased steam pressures at the inlet to the HP turbine will cause the unit to run with control valves in a more throttled position, thus increasing throttling losses across the valves. There will be a slight decrease in steam flow through the turbine leads, due to the increase in reheating steam to the moisture separator-reheater. This will have negligible effect on velocities and pressure drops in these lines. Since this is a small change in flow, stage pressures in the turbine will be approximately the same with the proposed return to NOP/NOT operations.

Due to the increased full power operating pressure in the steam generators, the feed pumps will require more power and the two feed pump turbines (FPT) will require more steam flow to power the two feed pumps, which will run at a higher speed. Higher temperature reheat steam to the FPT will be offset by its lower density, enabling the FPT to pass less mass. This results in little net change in FPT performance.

The main turbine speed/load is controlled via turbine control valves in one of two modes: speed control or megawatt (MW) control. When the turbine is in speed control mode, it is a feedback system, where the control valve is positioned based on current turbine speed. When the turbine is in MW control mode, the control valve is positioned based on the generator output. In either case, the proposed change in operating conditions will not adversely impact the system. When the generator is synchronized to the grid, a tunable constant is used to maintain the turbine's stability during and after the transition. Engineering personnel closely monitor the turbine during synchronization in order to tune the constant appropriately.

As a result, it was concluded that the main turbine and feed pump turbine are not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Moisture Separator Reheater (MSR)

The proposed changes in RCS operating conditions result in higher steam pressure in the main steam header; as a result, reheating steam flow to the reheater is increased. HP turbine exhaust flow to the moisture separator decreases by less than 1% at the proposed conditions. As a result, MSR performance will not be affected although the reheat temperature will be higher. Pressure drop decreases will be insignificant. Decreased main steam moisture carryover will tend to raise moisture separator effectiveness. This small increase will have little effect on the rest of the cycle.

As a result, it was concluded that the MSR is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Turbine Steam Seal System

The turbine steam seal system provides sealing steam at locations on the turbine shaft where it passes through the casing. Its purpose is two-fold: 1) to prevent steam from leaking out along the shaft and into the atmosphere, and 2) to prevent air from entering the turbine shells where vacuum in the cycle exists, e.g., at the exhaust of the LP turbines. Pressures and enthalpies at each packing or gland change less than 1% at the proposed conditions. As such, seal flows will remain approximately the same.

As a result, it was concluded that the turbine steam seal system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Main Steam System

This system consists of piping from the steam generator to the turbine, turbine bypass piping, steam generator stop valves, safety valves, and power operated relief valves. With the proposed change in RCS conditions, the full power main steam pressure will increase to approximately 800 psig at the steam generator outlet, with a saturation temperature of 520.4°F. Operation at the proposed conditions remains within the system and component design parameters of 1100 psig and 550°F, which encompass the more limiting no load steam conditions. Velocities in the steam leads will decrease, but remain in the recommended design velocity range.

Section 4.1 of WCAP-17762-NP, Rev. 1 (Enclosure 6 of this letter) documents an evaluation of plant control systems, which concluded that operation at the proposed conditions is bounded by existing control system analysis. This evaluation included consideration of pressure control component sizing.

Because components will be operating within their design limits and an evaluation has shown that safety valve and relief valve size requirements are not affected, it was concluded that the main steam system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Steam Condition Effect on the Turbine Missile Analysis

Because the HP turbine casing will contain any potential missiles, the Turbine Missile Analysis is based on a LP Turbine missile. The turbine missile analysis for the new LP turbines installed in 2011 is based on operating conditions that bound the proposed NOP/NOT conditions. As a result, it was concluded that the turbine missile analysis is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Ice Condenser Refrigeration System

The ice condenser refrigeration system cools down the ice condenser from containment ambient conditions and maintains the desired equilibrium temperature in the ice compartment. The ice condenser is sufficiently subcooled and insulated so that even a complete breakdown of the refrigeration or air handling system would not cause ice melting for one week.

The ice condenser refrigeration system consists of 10 glycol chiller units, 6 glycol circulation pumps and 60 air handling units for each containment building. The allowable containment temperatures during operation in Technical Specifications 3.6.5, Containment Air Temperature, and 3.6.11, Ice Bed, are not changed by the proposed return to NOP/NOT conditions.

As a result, it was concluded that the ice condenser refrigeration system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Emergency Core Cooling System (ECCS)

The ECCS is one of the four engineered safety feature systems that mitigate the consequences of a major rupture of the RCS or main steam system pipes inside containment. The ECCS consists of six ECCS pumps [two high head centrifugal charging (CC) pumps, two medium head safety injection (SI) pumps, and two low head residual heat removal (RHR) pumps, two heat exchangers, four accumulator tanks, and associated piping, valves, and instrumentation.

During accident mitigation, the ECCS provides significant mass injection to the RCS for volume makeup as well as core cooling and reactivity control. The source of borated makeup water to the ECCS pumps is the refueling water storage tank (RWST). In addition, accumulators provide passive injection of borated water to the RCS loops as the RCS pressure drops below accumulator pressure.

ECCS operation in response to a LOCA occurs in two phases. The Injection Phase begins upon receipt of a safety injection signal and results in automatic starting of the ECCS pumps to transfer RWST contents to the RCS to provide makeup for lost coolant and core cooling/reactivity control. As the RWST contents are being depleted, the ECCS pumps are stopped one train at a time and their suction is realigned from the RWST to the containment recirculation sump. This realignment ends the ECCS Injection Phase and begins the ECCS Recirculation Phase, the latter of which provides long term reactor core and containment cooling.

The proposed increase in normal RCS operating pressure and full power T_{avg} values primarily affects normal operating conditions and not post-accident mitigation functions. The proposed changes do not include changes in reactor thermal power; therefore, decay heat loads that must be removed in response to postulated accident mitigation will not change.

Proposed changes to the settings of the time delay relays in the CTS pump and the Containment Air Recirculation/Hydrogen Skimmer (CEQ) System fan starting circuits, which are being made to support BELOCA-PCT analysis, do affect post-accident ice melt rates and can impact minimum containment sump water level during the ECCS Recirculation Phase. In addition to BELOCA-PCT and minimum sump water level, these relay setting changes can influence the achievement of accident analyses acceptance criteria for steam line break inside containment, LOCA and steam line break containment integrity, and control room habitability and offsite dose analyses.

The effects of the proposed NOP/NOT Program changes identified in the previous paragraph on the Unit 1 accident analyses were addressed by Westinghouse and are described in Sections 5.1, 5.4 and 5.5 of WCAP-17762-NP, Rev. 1 (Enclosure 6 of this letter). Impacts on the control room habitability and offsite dose consequence analyses were addressed by I&M and are documented in Enclosure 8 of this letter. Finally, impacts on minimum containment sump water level were evaluated by Fauske & Associates for Cook Plant. In all cases, the accident analyses acceptance criteria pertinent to Enclosures 6 and 8 of this letter continue to be met with implementation of NOP/NOT-related time delay relay setting changes in the starting circuits for the CTS pumps and CEQ fans. In addition, it was determined that the proposed NOP/NOT changes will maintain the licensing basis minimum containment recirculation sump water level during ECCS Recirculation.

As a result of the reviews described in the previous paragraph, it was concluded that the emergency core cooling system is not adversely affected by the proposed increase in RCS normal operating pressure and average full power temperature, and by the proposed changes in the time delay relay settings in the CTS pump and CEQ fan starting circuits.

Equipment and Floor Drainage System

The purpose of the equipment and floor drainage system is to collect and dispose of liquids from floors and equipment. The NOP/NOT project will not impact the floor drain flow. As a result, it was concluded that the equipment and floor drainage system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Fire Protection Systems and Fire Hazards Analysis

The fire protection systems and Fire Hazards Analysis are independent of RCS operating characteristics. As a result, it was concluded that the fire protection systems and Fire Hazards Analysis are not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

HVAC Systems

The HVAC systems in the auxiliary building, screenhouse, and containment were reviewed to determine the impact of NOP/NOT operation. Since the RCS temperature and main steam temperature will increase with NOP/NOT, slightly more heat will be lost from the RCS and main steam lines. However, the HVAC systems will not be adversely affected, since the thermal heat loads do not significantly increase and are within the original design conditions for these systems.

Chemical and Volume Control System (CVCS)

The CVCS provides reactivity control by regulating the concentration of boric acid solution neutron absorber in the RCS. Reactivity control is provided through the addition and removal of boric acid through the boron makeup system of CVCS. The proposed increases in normal RCS operating pressure and full power average temperature do not affect the reactivity control function of the CVCS,

The proposed increase in normal RCS full power average temperature does, however, impact the pressurizer level control program, which could affect CVCS normal operations. During unit loading and unloading operations, the reactor vessel average temperature (T_{avg}) changes to follow the T_{avg} program. The result is a swell or shrink of the water volume in the RCS, which is observed as a change in the pressurizer water level. The function of the pressurizer level control system is to maintain the pressurizer level at or near its programmed level as a function of measured Tavg. The goal of the pressurizer level program is to maintain an approximate constant mass inventory in the RCS during load changes so that the CVCS charging rate can remain relatively constant. To address the proposed full power average temperature increase to 571°F, the pressurizer water level control program will be revised, consistent with existing Westinghouse design basis, to satisfy the program's primary goal of maintaining a relatively constant CVCS charging rate.

As a result, it was concluded that the chemical and volume control system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Turbine Bypass System

The turbine bypass system (steam dump) is a control system that allows steam to bypass the turbine and go directly to the condenser. The steam dump system is physically sized to provide the capacity of 26% to 39% of full load steam flow, depending on steam pressure.

An evaluation of the Nuclear Steam Supply System design transients, which includes consideration for turbine bypass system operation, was performed to determine the continued applicability of the design transients assuming a return to NOP/NOT conditions. This evaluation, which is described in Section 3.1 of Enclosure 6 of this letter, found that the current design transients support an NSSS power level up to 3600 MWt, a full power vessel average temperature (T_{avg}) window from 547°F to 581.3°F, operating pressure of either 2000 psia or 2250 psia, and steam generator tube plugging (SGTP) level of up to 30 percent. These conditions bound the proposed RCS operating conditions defined in this License Amendment Request.

As a result, it was concluded that the turbine bypass system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Radiation Monitoring System

Since reactor thermal power is not changing as part of the return to RCS normal operating pressure and full power average temperature, the source term is unchanged. It was concluded, therefore, that the radiation monitoring system is not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Radioactive Waste Treatment and Waste Disposal Systems

Liquid, gaseous, and solid waste disposal facilities are designed so that discharge of effluents and off-site shipments are in accordance with applicable governmental regulations. Sizing of various waste handling and processing equipment was predicated on the volumes and flow rates originally expected to be handled. The waste disposal system design is based on continuous operation of the primary plant with one percent defective fuel. Because reactor thermal power is not changing under the proposed return to NOP/NOT conditions, the source term and quantities of wastes are unchanged. It was concluded, therefore, that the radioactive waste treatment and disposal systems are not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Secondary Systems Piping Supports

The secondary systems' piping support requirements were designed for process conditions that encompass the proposed NOP/NOT conditions. It was concluded, therefore, that the secondary systems' piping supports are not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Electrical Systems

Since the overall heat balance is not significantly changed, there is no adverse impact on the Balance of Plant electrical systems. One change to plant loads as a result of the proposed NOP/NOT program is associated with the reactor coolant pumps (RCPs). Required power for the RCPs will be decreased slightly due to the decreased water density at raised T_{avg} conditions. Finally, the proposed changes to the time delay relay settings in the starting circuits for the CTS pumps and CEQ fans will not adversely affect loading on the emergency diesel loading since the time delay settings are being increased. It was concluded, therefore, that the electrical systems are not adversely affected by the proposed increase in RCS normal operating pressure and full power T_{avg}.

Enclosure 8 to AEP-NRC-2013-79

Radiological Dose Evaluation

Radiological Dose Evaluation

1.0 Introduction and Background

Offsite and control room habitability radiological dose consequence analyses were assessed for possible impacts from the proposed Unit 1 Return to Normal Operating Pressure/Normal Operating Temperature (NOP/NOT) Program. In addition to reviewing accident-specific dose analyses, associated calculations for core source terms, reactor coolant system (RCS) source terms, and atmospheric dispersion factor were included in the assessment.

2.0 Input Parameters and Assumptions

The following key parameters, which are being revised for the Return to NOP/NOT Program, were reviewed as part of the effort discussed herein:

1. Nominal pressurizer pressure from 2100 psia to 2250 psia.
2. Vessel average temperature (T_{avg}) at full power from 556°F to 571.0°F.
3. Containment Air Recirculation / Hydrogen Skimmer (CEQ) fan actuation delay time from 180 seconds to 300 seconds.
4. Containment Spray (CTS) actuation delay time, assuming a loss of offsite power, from 180 seconds to 300 seconds.

3.0 Description of Evaluation

Overview

Offsite and control room habitability radiological dose consequence analyses and the associated supporting calculations were examined for potential adverse impacts from the proposed Return to NOP/NOT Program. The examination was conducted in two parts.

The first part consisted of 1) an initial screening of specific dose analyses to determine if design inputs included CEQ fan delay time, CTS pump delay time, RCS operating temperature, RCS operating pressure, or RCS mass [which is affected by changes in RCS temperature and pressure], and 2) a review of any of the dose analyses found to use such inputs to determine if the proposed NOP/NOT parameter and system changes remained bounded by the current dose analyses of record.

The second part of the impact examination consisted of a detailed review of specific dose analyses that use design inputs that would not bound the proposed return to NOP/NOT. The detailed review consisted of sensitivity analyses that quantified the impacts of the proposed NOP/NOT changes on the applicable dose analyses.

Screening Assessment of RCS Pressure, Temperature and Mass Changes

The proposed increase in Unit 1 normal operating RCS pressure to 2500 psia and full power average temperature to 571°F impacts RCS mass, which is used as a design input throughout the dose consequence analyses and supporting calculations. Since the dose consequence analyses of record are common to both Donald C. Cook Nuclear Plant Units 1 and 2, bounding mass values are used. Given that Unit 1 has a greater RCS liquid volume than Unit 2 and current Unit 1 RCS conditions have a higher density than Unit 2, current Unit 1 parameters are used to establish maximum RCS liquid mass and Unit 2 parameters are used to establish the minimum RCS liquid mass.

Regarding the changes in RCS temperature to 571.0°F and pressure to 2250 psia for Unit 1 NOP/NOT, RCS fluid density will decrease, thus affecting the previous maximum RCS liquid mass value used in some of the dose consequence analyses. As a result, the existing dose analyses which conservatively use maximum RCS mass will remain bounding. At the other end of the RCS mass spectrum, the minimum RCS mass used in some dose analyses is based on Unit 2 RCS liquid volumes, and RCS T_{avg} and pressure values of 574°F and 2250 psia, respectively. Compared to the proposed Unit 1 NOP/NOT values, the Unit 2 conditions remain bounding for minimum RCS mass because they continue to have a lower RCS fluid density and a smaller fluid volume.

As a result of the above screening review of the proposed RCS pressure, temperature, and mass changes, it was determined that the current offsite and control room habitability dose analyses of record remain bounding and are not affected by the proposed return to NOP/NOT conditions for Unit 1.

Screening Assessment of Proposed Changes to CTS and CEQ Fan Actuation Times

A thorough screening of the offsite and control room habitability radiological dose consequence analyses and supporting calculations determined that only the LBLOCA dose analyses required further evaluation due to proposed changes in CTS and CEQ fan actuation times. The remaining dose analyses are not affected since they do not model CTS or CEQ fan actuation.

4.0 LBLOCA Dose Analyses Acceptance Criteria

The control room habitability dose consequence analyses utilize the Alternative Source Term (AST) methodology of Regulatory Guide 1.183 (Reference 1). The acceptance criterion for this LBLOCA analysis is 5 rem TEDE for the duration of the accident being analyzed. This limit originates in 10 CFR 50.67 (Reference 2).

The LBLOCA offsite radiological dose consequence analyses are based on the Technical Information Document (TID)-14844 (Reference 3) methodology described in Regulatory Guide 1.195 (Reference 4). The acceptance criteria for the Large Break Loss of Coolant Accident analyses come from Table 4 of Reference 4.

5.0 Evaluation Results

Explicit modeling of the increased CTS and CEQ delay times at NOP/NOT conditions resulted in a negligible impact on the LBLOCA control room habitability radiological dose consequence analysis. The acceptance criterion of 5 rem TEDE for the duration of the accident from 10 CFR 50.67 continues to be met.

The increased CTS and CEQ delay times had a greater effect on the offsite radiological dose analyses, but the acceptance criteria from Table 4 of Regulatory Guide 1.195 continue to be met. Additionally, the increase in dose was less than 10 percent of the difference between the current calculated dose values and the regulatory guideline values of 300 rem thyroid and 25 rem whole body for both the Exclusion Area Boundary and Low Population Zone. Therefore, the change in analytical results is considered to be minimal.

6.0 Conclusions

It is concluded that the Return to Unit 1 NOP/NOT conditions has a minimal impact on the offsite and control room habitability radiological dose consequence analyses and the associated supporting calculations. The acceptance criteria from 10 CFR 50.67 for control room habitability and Regulatory Guide 1.195 for offsite dose continue to be met when considering the proposed input parameter changes. Additionally, for those analyses affected by the Return to NOP/NOT conditions, the increase in dose is not significant since it is less than 10 percent of the difference between the current calculated dose values in the analyses of record and the regulatory guideline values.

7.0 References

1. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.
2. 10 CFR 50.67, "Accident Source Term," December 1999.
3. J.J. DiNunno et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, United States Atomic Energy Commission, March 1962.
4. Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," May 2003.

Enclosure 9 to AEP-NRC-2013-79

Expected LOCA Peak Clad Temperature Summary

Westinghouse LOCA Peak Clad Temperature Summary for ASTRUM Best Estimate Large Break

Plant Name: Donald C. Cook Unit 1
Utility Name: American Electric Power
Revision Date: 8/27/2013

Analysis Information

EM:	ASTRUM (2004)	**Analysis Date:**	11/20/2007	**Limiting Break Size:**	Split
FQ:	2.15	**FdH:**	1.55		
Fuel:	15x15 Upgraded	**SGTP (%):**	10		
Notes:	Post-Analysis evaluation for FQ of 2.09 and FdH of 1.53				

	Clad Temp (°F)	Notes
LICENSING BASIS		
Analysis-Of-Record PCT	2128	
PCT ASSESSMENTS (Delta PCT)		
A. PRIOR ECCS MODEL ASSESSMENTS		
1 . Update to LOTIC2 Calculated Containment Pressure	0	
B. PLANNED PLANT MODIFICATION EVALUATIONS		
1 . Design Input Changes with Respect to Plant Operation for Return to NOP/NOT Evaluation	-489	(a)
C. 2013 ECCS MODEL ASSESSMENTS		
1 . Return to NOP/NOT Including Pellet Thermal Conductivity Degradation and Peaking Factor Burndown	404	(a)
2 . Revised Heat Transfer Multiplier Distributions	-91	(b)
D. OTHER		
1 . None	0	
LICENSING BASIS PCT + PCT ASSESSMENTS PCT =	1952	

References:

1. AEP-NRC-2012-13, "Donald C. Cook Nuclear Plant Units 1 and 2 Response to Information Request Pursuant to 10 CFR 50.54(f) Related to the Estimated Effect on Peak Cladding Temperature Resulting from Thermal Conductivity Degradation in the Westinghouse-Furnished Realistic Emergency Core Cooling System Evaluation (TAC No. M99899)," March 19, 2012.
2. NRC Letter, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Evaluation of Report Concerning Significant Emergency Core Cooling System Evaluation Model Error Related to Nuclear Fuel Thermal Conductivity Degradation (TAC Nos. ME8322 and ME8323)," March 7, 2013
3. AEP-NRC-2013-68, "Donald C. Cook Nuclear Plant Unit 1 30-Day Report of Changes to or Errors in an Evaluation Model," August 30, 2013.

Notes:

(a) The original TCD evaluation for Unit 1 in Reference 1 is superseded by the return to Unit 1 NOP/NOT evaluation. NRC review of Reference 1 is documented in Reference 2.

(b) The original revised heat transfer multiplier distribution line item transmitted to NRC in Reference 3 is superseded by the revised heat transfer multiplier distribution line item at NOP/NOT conditions.