H.B. Robinson / Shearon Harris Transient Analysis Methodology

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NRC Offices

Presentation Outline

- Update on Proposed Submittals
- Licensing Approach
- Overview of DPC-NE-3009-P
- Scope of Demonstration Analyses
- Core Physics and Power Distribution Analysis Methodology
- Methodology Changes
- Conclusion

Methods Reports

	MNS/CNS	ONS	Proposed RNP/HNP	Submittal Date
Physics Codes / Models	DPC-NE-1005 CASMO-4/SIMULATE-3	DPC-NE-1006 CASMO-4/SIMULATE-3	DPC-NE-1008 CASMO-5/SIMULATE-3	**August 19, 2015**
Physics Applications Power Distribution Monitoring	DPC-NE-2011	NFS-1001 DPC-NE-1002	DPC-NE-2011 revision	**February 3, 2016** **Withdrawn April 7, 2016** Resubmit April/May 2016
Physics Applications Reload Design	DPC-NF-2010	NFS-1001 DPC-NE-1002	DPC-NF-2010 revision	**February 3, 2016** **Withdrawn April 7, 2016** Resubmit April/May 2016
NSSS Codes / Models	DPC-NE-3000 RETRAN-02	DPC-NE-3000 RETRAN-3D	DPC-NE-3008 RETRAN-3D	**November 19, 2015**
Subchannel T/H Methods	DPC-NE-3000 DPC-NE-2004 VIPRE-01	DPC-NE-3000 DPC-NE-2003 VIPRE-01	DPC-NE-3008 DPC-NE-2005 (Appendix) VIPRE-01	**November 19, 2015**
SCD Methodology	DPC-NE-2005	DPC-NE-2005	DPC-NE-2005 revision	**March 5, 2015** **Approved March 8, 2016**
Transient Analysis	DPC-NE-3001 DPC-NE-3002 SIMULATE-3K (REA)	DPC-NE-3005 SIMULATE-3K (REA)	DPC-NE-3009 SIMULATE-3K (REA)	June/July 2016
Fuel Performance	DPC-NE-2008 (TACO-3) DPC-NE-2009 (PAD 4.0)	DPC-NE-2008 (TACO-3 and GDTACO)	N/A - TS changes only COPERNIC-2 / PAD-5	December 2016

Licensing Approach

- DPC-NE-3009-P describes the modeling approach used in performing the (U)FSAR Chapter 15 non-LOCA transient analysis
- Extends the Duke DPC-NE-3001-PA / DPC-NE-3002-PA methodology to the Harris and Robinson Nuclear Plants
- System response uses the RETRAN-3D computer code and models described in DPC-NE-3008-P (submitted to the NRC 11/2015)
- DNBR analysis will use the VIPRE-01 models described in DPC-NE-2005-P (approved by the NRC 3/2016) or the extended VIPRE-01 models described in DPC-NE-3008-P

Licensing Approach (cont.)

- LAR submittal
  - Intend to submit DPC-NE-3009 as a supplement to the DPC-NE-3008 LAR to avoid linked submittal concerns
  - References to DPC-NF-2010 will be provided as an example method that can be used (or an equivalent NRC-approved method)
  - Tech Spec 5.6.5 and 6.9.1.6 changes

- (U)FSAR changes
  - Implemented via 10 CFR 50.59 following methodology report approval with first in-house reload analysis

Schedule

- Support the reload licensing analysis for Harris Cycle 22 and Robinson Cycle 32
  - H1EOC21 (4/18)
  - R2EOC31 (9/18)
- Reload Analyses Start:
  - HNP (December 2016)
  - RNP (Spring 2017)
- Review requested by the middle of 2017

Overview of DPC-NE-3009-P

- Defines the method used to analyze non-LOCA (U)FSAR Chapter 15 accidents and transients
- Establishes the analysis and modeling assumptions to bound the licensed operating conditions for the current plant design and fuel cycle
- Establishes a set of key physics parameters important to each accident that are verified on a cycle-specific basis
- Methodology used to demonstrate that accident acceptance criteria are satisfied
  - Fuel design limits
  - System overpressure design limits
  - Provides input to dose analysis used to confirm acceptability of dose consequences

Overview of DPC-NE-3009-P (cont.)

- 1.0 Introduction
- 2.0 Background
- 3.0 Simulation Codes and Models
- 4.0 Safety Analysis Physics Parameters

Overview of DPC-NE-3009-P (cont.)

- 5.0 Transient Analysis Methods
  - 5.1 Increase in Heat Removal by the Secondary System
  - 5.2 Decrease in Heat Removal by the Secondary System
  - 5.3 Decrease in Reactor Coolant System Flow Rate
  - 5.4 Reactivity and Power Distribution Anomalies
  - 5.5 Increase in Reactor Coolant System Inventory
  - 5.6 Decrease in Reactor Coolant System Inventory
- 6.0 Demonstration Analyses
- 7.0 Summary

Scope of Demonstration Analyses

- Demonstration analyses completed to provide sample results for various events
- Input assumptions derived from various sources
  - Current design inputs, preliminary nuclear design models, proposed input changes, etc.
- Demonstration analyses not intended for direct incorporation into the (U)FSAR
- Demonstration analyses not being submitted for review and approval as new AORs

Scope of Demonstration Analyses (cont.)

Scope of Demonstration Analyses		Plant/Event Evaluated	
		Harris	Robinson
15.1	Increase in Heat Removal by the Secondary System	Steam System Piping Failure (MSLB)	Steam System Piping Failure (MSLB)
15.2	Decrease in Heat Removal by the Secondary System	Turbine Trip (Submitted for review in separate LAR)	
15.3	Decrease in Reactor Coolant System Flow Rate	Loss of Flow Locked Rotor	
15.4	Reactivity and Power Distribution Anomalies	RCCA Ejection	Uncontrolled Bank Withdrawal at Power Withdrawal of a Single Full-Length RCCA RCCA Ejection

Core Physics and Power Distribution Analysis Methodology

- Example analysis methodology is based on the following report:
  - DPC-NF-2010, *Nuclear Physics Methodology for Reload Design*
- DPC-NF-2010 describes Duke's overall reload design methodology and the methodology used to calculate core physics parameters
- An NRC-approved methodology is used to calculate core physics parameters

Physics Codes

- CASMO-4/CASMO-5: Develops few-group nuclear data for each unique fuel assembly region in the core
  - Cross sections
  - Kinetics data
  - Assembly discontinuity factors
  - Fission product data
- SIMULATE-3: Three-dimensional steady-state core simulator used to calculate reactivity and core power distributions and perform fuel depletion
- SIMULATE-3K: Three-dimensional transient core simulator used for analysis of the RCCA ejection accident
  - Model has all of the capabilities of SIMULATE-3 but explicitly accounts for the time-dependent behavior of the neutronics and thermal-hydraulics of each fuel assembly in the reactor core

Key Physics Parameters

- Calculation methodology follows approach developed in DPC-NE-3001-PA
- Methodology defines parameters that are important to the transient response for each accident
- Selection objectives
  – Bound expected reload values at nominal and transient conditions
  – Retain margin to account for uncertainty and future core design and/or operational changes
  – Minimize impact on reload core design
  – Minimize the potential for re-analysis

Parameter Validation

- Cycle-specific confirmation is performed for each reload core
- Parameters are calculated assuming a conservative set of initial conditions and compared against the accident analysis assumptions
- Reload calculated values consider the impacts of:
  - Power level
  - Soluble boron concentration
  - Control rod position
  - Burnup
  - Xenon
- If a non-conservative parameter is found during verification of the safety analysis, then the accidents affected must be re-evaluated or the loading pattern revised

Thermal Limits Evaluation

- Core power distributions are compared against departure-from-nucleate-boiling ratio (DNBR) and centerline fuel melt (CFM) limits to confirm accident analysis acceptance criteria are satisfied
- For accidents where DNB and CFM are allowed to occur, the number of fuel rods exceeding their limit is confirmed less than the value assumed in the dose analysis
- Appropriate uncertainties and penalties are applied to the calculated peaking factors prior to comparison to DNBR and CFM limits

Selected Methodology Changes (Relative to DPC-NE-3001-PA and DPC-NE-3002-A)

- **Main Steam Line Break**
  - RETRAN-3D nodalization changes from DPC-NE-3008 to model Main Steam Line Break (MSLB)
  - CHF correlations

- **RCCA Ejection**
  - Gap closure model
  - Initial gap conductance
  - Enthalpy calculation

Overview of the MSLB Analysis

- Event is initiated from a break in the main steam line
- Depressurization of the steam generator produces a rapid reactor coolant system cooldown and depressurization
- Reactivity insertion from the cooldown (due to negative MTC) causes a return to power, and if a stuck rod is assumed, results in high peaking factors which could potentially challenge DNBR and CFM limits
- System analysis is performed with RETRAN-3D using a split reactor vessel model
- Physics inputs are from an NRC-approved nuclear physics method (such as DPC-NF-2010)

Overview of the MSLB Analysis (cont.)

- Reactivity addition in RETRAN-3D is verified conservative by modeling the core conditions at the minimum DNBR state point in SIMULATE-3
  - SIMULATE-3 must be subcritical at the RETRAN-3D state point conditions
- Multi-channel VIPRE-01 model is used with the power distribution at the peak heat flux state point to confirm DNBR is above the CHF limit

MSLB Split Reactor Vessel Model

MSLB Secondary System Modeling (HZP Cases)

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MSLB Core Physics Model

- Asymmetric inlet temperature and flow conditions are modeled based on the orientation of the affected loop
- The stuck rod is assumed in the affected region of the core to maximize peaking
- The SIMULATE-3 reactivity prediction is used to verify that the RETRAN kinetics model is conservative
- SIMULATE-3K is used for LOOP case to account for voiding
- SIMULATE-3 power distributions are input to VIPRE-01 to verify the DNBR acceptance criterion is met

MSLB VIPRE-01 Model

MSLB CHF Correlations

RCCA Ejection Analysis

- Accident is initiated by a failure in the control rod drive mechanism housing
- The control rod is rapidly ejected from the reactor coolant system pressure boundary
- The resulting power excursion is a function of the reactivity worth of the ejected rod
- The key reactor protection trips are:
  - High Neutron Flux
  - High Neutron Flux Positive Rate
  - Over-Power and Over-Temperature Delta-T

RCCA Ejection Analysis (cont.)

- SIMULATE-3K is used to model the transient power response and the high flux and flux rate trips
- Bounding physics parameters are assumed
- Determined some low ejected rod worth cases do not trip on high flux or high flux rate
  - ΔT trips must be credited
- Causal factors Include:
  - Orientation of control bank rods relative to the excore detectors
  - Lack of a high flux positive rate trip (RNP only)
- Event looks like a single rod withdrawal accident except with a hole in the reactor pressure vessel
  - RETRAN-3D used to analyze the transient using the power response from SIMULATE-3K

Example Robinson HFP REA Neutron Power vs. Time

RCCA Ejection Analysis (cont.)

RCCA Ejection Analysis (cont.)

Conclusion

- DPC-NE-3001/3002 Transient Analysis method modified for use at HNP and RNP
- Minor changes include:
  - Computer code upgrades (RETRAN-3D)
  - RETRAN nodalization consistent with models in DPC-NE-3008
  - REA changes to address plant-specific differences (RETRAN-3D modeling of ΔT trips)
  - REA modeling enhancements (VIPRE-01 dynamic gap model, SIMULATE-3K enthalpy calculation)
  - Added MSLB CHF correlations for conditions outside HTP correlation range (Modified Barnett, EPRI)
- Selected demonstration calculations provided to illustrate the application of the methods to HNP and RNP