UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

February 6, 1992

Docket No. 50-482

LICENSEE: Wolf Creek Nuclear Operating Corporation

FACILITY: Wolf Creek Generating Station

SUBJECT: SUMMARY OF MEETING HELD ON JANUARY 28, 1992 REGARDING TOPICAL REPORTS SUBMITTED BY WOLF CREEK NUCLEAR OPERATING CORPORATION

On January 28, 1992, members of the NRC staff met with representatives of Wolf Creek Nuclear Operating Corporation to discuss the ongoing reviews of nuclear design and transient analysis topical reports. A list of the attendees is provided as Enclosure 1. Those meeting handouts which do not contain proprietary information are provided as Enclosures 2 through 5.

The first discussion involved the proposed schedule for topical report and Technical Specification revision submittals to the NRC. A total of five topical reports are planned by the licensee to support the inhouse performance of core reload design, transient analysis, and an increase in the rated thermal power of the Wolf Creek Generating Station (WCGS). The NRC review of the following topical reports is ongoing:

1) Rod Exchange Methodology for Startup Physics Testing
2) Transient Analysis Methodology for WCGS
3) Core Thermal-Hydraulic Analysis Methodology for WCGS

Based upon discussions during and after the meeting, the expected completion schedules for the NRC review of the above topicals range from Spring 1992 to November 1992. These schedules are adequate to support the licensee's design of the Cycle 7 reload (startup from refueling outage scheduled for Spring 1993) and the power uprate submittal expected in Summer 1993.

In addition to the above topical reports, topical report "Qualification of Steady State Core Physics Methodology for Wolf Creek Design and Analysis" was submitted on January 15, 1992 and the Reload Safety Evaluation Methodology topical report is scheduled for submittal in March 1992. NRC review of these topicals would also be required to support the licensee's design of Cycle 7 and the power uprate. The licensee was made aware of existing NRC budget problems related to topical report reviews and the potential impact on the review of the nuclear design topical. At the time of the meeting, the schedules related to the NRC review of these topicals has not been established. The licensee's proposed schedule is provided as Enclosure 2. A summary of the status of the licensee's power uprate program is provided as Enclosure 3.

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Detailed discussions related to the review of the rod exchange topical and the transient analysis topical were held between the licensee and NRC staff from the Reactor Systems Branch. The licensee plans to submit the requested information associated with the rod exchange topical in February 1992. A preliminary set of questions from the NRC contractor reviewing the transient analysis was distributed at the meeting and is provided as Enclosure 4. The licensee plans to review the questions and establish a schedule for a conference call with the contractor and NRC staff. A handout (Enclosure 5) associated with the licensee's approach to responding to an NRC request for additional information regarding the thermal-hydraulic topical was discussed briefly.

Original Signed By

William D. Reckley, Project Manager
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Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. List of Attendees
2. Licensee Handout
   Proposed Schedule
3. Licensee Handout
   Power Uprate Program
4. Draft Request for
   Additional Information
5. Licensee Handout
   Approach to T/H Topical RAI

cc: See next page

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AHowell, RGN-IV
MChatterton
CYLiang
FOrr
JThomas
SBlack
NRC Participants
SShankman

*See Previous Concurrence Page

OFC	: PDIV-2/LA*	: PDIV-2/PM	: PDIV-2/D	:	:
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DATE	: 2/3/92	: 2/4/92	: 2/4/92	:	:

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ENCLOSURE 1

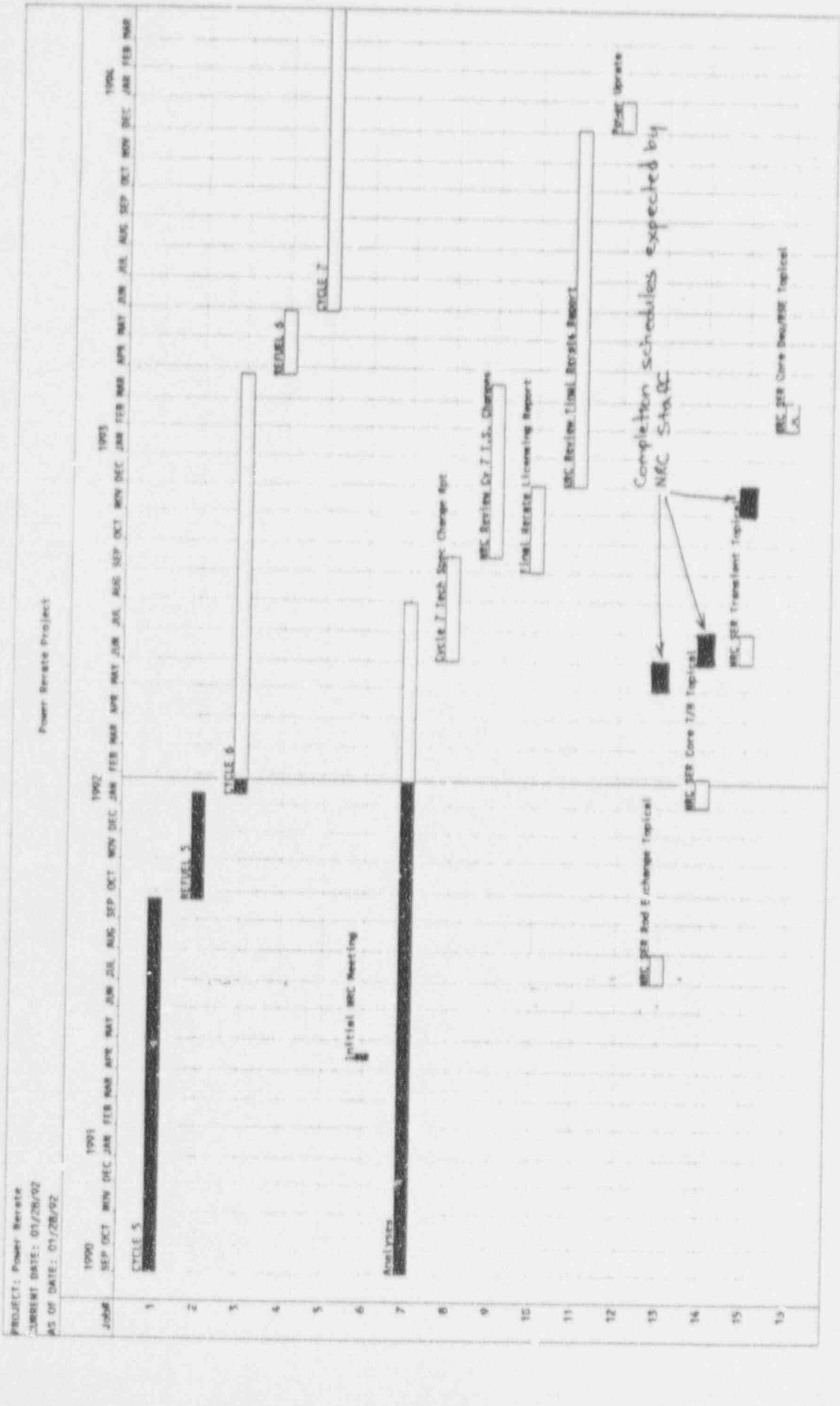
List of Attendees

Wolf Creek Nuclear Operating Corporation

Steve Wideman, Supervisor- Licensing
Terry Garrett, Manager- Nuclear Safety Analysis
Elliot Jackson, Supervisor- Core Design
Glenn Neises, Engineer- Nuclear Safety Analysis

NRC

William Reckley, NRR/PD42
Margaret Chatterton, NRR/SRXB
Chu-Yu Liang, NRR/SRXB
Frank Orr, NRR/SRXB
John Thomas, NRR/SRXB


PROJECT: Power Rerate
CURRENT DATE: 01/28/92
AS OF DATE: 01/28/92
Power Rerate Project
1990
1991
1992
1993
1994
CYCLE 5
REFUEL 5
CYCLE 6
REFUEL 6
CYCLE 7
Initial NRC Meeting
Analyses
Cycle 7 Tech Spec Change Rpt
NRC Review Cy 7 T.S. Changes
Final Rerate Licensing Report
NRC Review Final Rerate Report
Power Uprate
NRC SER Rod Exchange Topical
NRC SER Core T/H Topical
NRC SER Transient Topical
Completion schedules expected by NRC Staff

ENCLOSURE 3

WOLF CREEK NUCLEAR OPERATING CORPORATION

POWER RERATING PROGRAM STATUS

PHASE 3 REVIEW

JANUARY 28, 1992

POWER RERATING PROGRAM PHASES

- Phase 1 WCGS Scoping Study
  - Plant Target Configuration

- Phase 2 Technical Basis Development
  - Assumptions
  - Program Basis

- Phase 3 Limiting Events Evaluation
  - Limiting Events Analysis
  - ID Plant Modifications

- Phase 4 Analysis and Documentation
  - Non-Limiting Events Evaluations & Analyses

- Phase 5 Implementation
  - Tech Spec / USAR Changes
  - Final Report
  - NRC Review

POWER RERATE
PLANT TARGET CONFIGURATION

ITEM	TARGET VALUE
Increase Core Thermal Power	3565 MWt
Reduction in T_{HOT}	Operation 5 ^{o}F Analysis 15 ^{o}F
Intermediate Flow Mixers	Addition
$F_{\Delta H}$ / F_Q	1.6[illegible] / 2.5
Positive Moderator Temperature Coefficient	+7 pcm/^{o}F
LOCA Re-Analysis	BASH Model
Thimble Plugs	Removal
Allowed Steam Generator Tube Plugging	10%
Negative Flux Rate Trip	Elimination
Secondary Side Safety Injection Trip	Elimination
Core Operating Limit Report	Preparation

RERATE PROGRESS

Core Thermal-Hydraulics

- Technical Basis Development Scoping Studies
- Core Thermal-Hydraulic Model Sensitivity Studies
- Confirmation of Plant Target Configuration
- Limiting Events DNB Analysis

Safety Analysis

- Transient Model Changes to Incorporate Plant Target Configuration
  - PMTC
  - SI/AFW Flowrates
  - OTDT/OPDT Setpoint Optimization

- Calculation of Revised ECCS / AFW Flows

- Containment Pressure / Temperature Analysis for Uprated Conditions

Safety Analysis

- Analysis of Limiting Events
    - Complete Loss of Forced Reactor Coolant Flow
    - Reactor Coolant Pump Shaft Seizure
    - Reactor Coolant Pump Shaft Break
    - Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power
    - Rod Cluster Control Assembly Misoperation
    - Spectrum of Rod Cluster Control Assembly Ejection Accidents

Core Design

- Technical Basis Development Review of Plant Target Configuration
- Generated Cross-Sections for the Plant Target Configuration
- Preliminary Estimates of Batch Size & Enrichment
- Development of Reload Design Interface Procedure
- Completion of WCNOC Core Design Methodology Report
- Completion of WCNOC Reload Safety Evaluation Report
- Cycle 7 Core Design (in progress)
- Core Operating Limit Report (in progress)

OTHER ITEMS

- Evaluation of Balance-of-Plant Systems

AB	Main & Reheat System
AC	Main Turbine
AD	Condensate
AE	Feedwater
AF	Feedwater Heater Extraction & Drains
AK	Condensate Demineralizer
CA	Steam Seals
CG	Condenser Air Removal
FC	Auxiliary Turbines (Feedpump Fan, etc)

- Evaluation of Transformer Changes to Increase Cooling

- Limiting Small & Large Break LOCA Analysis (in progress)

- NSSS Systems & Components Analysis

- Fluid & Control Systems Analysis

CLOSING REMARKS

- Submit a Cycle 7 Technical Specification Change Package Report: September 1992
- Obtain NRC Review / Approval of T.S. Changes by March 1993
- Submit Final Rerate Report: December 1992
- Obtain Final NRC Review / Approval for Power Increase: December 1993
- Increase Power: December 1993/January 1994

REPORT SUBMITTAL / REVIEW SCHEDULE

	WCNOC Submittal	Requested NRC Review/SER
Rod Exchange Topical	8/90	7/91
Core Thermal-Hydraulic Analysis Methodology Topical	8/90	1/92
Transient Analysis Methodology Topical	2/91	6/92
Core Physics Methodology Topical	1/92	2/93
Reload Safety Evaluation Methodology Topical	3/92	2/93
Cycle 7 T.S. Change Package	9/92	3/93
Power Rerating Report	12/92	12/93

* Letter No. ET 91-0026, from F. T. Rhodes (WCNOC) to USNRC, "Transient Analysis Methodology Topical", 2/1/91.

PROPOSED TECHNICAL SPECIFICATION CHANGES SUBMITTAL AND REVIEW SCHEDULE

Proposed Technical Specification Change	WCNOC NRC Submittal	NRC Approval
Definition of Rated Thermal Power	12/92	12/93
Cycle 7 Technical Specification Change Report		
WRB-2 CHF Correlation	9/92	3/93
$F_{\Delta H}$ / F_Q	9/92	3/93
PMTC	9/92	3/93
Negative Flux Rate Trip Elimination	9/92	3/93
Low Steam Line Pressure SI Elimination	9/92	3/93
Core Operating Limit Report	9/92	3/93

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ENCLOSURE 4

ATTACHMENT 1

Request for Additional Information
Review of Wolf Creek RETRAN Model Qualification

Part 1

1.0 General Approach

1. Justify thoroughly: (i) the plant nodalization on a transient-by-transient basis; and (ii) the variables (and values obtained) which were permitted to be adjusted by the RETRAN initialization routine. Furthermore, provide and describe in depth all parametric studies performed to lead WCNOC to conclude that the nodalization presented in Section 2.1 is either a best-estimate or a conservative representation of the plant, and demonstrate that use of code models is conservative.

2. In modeling the steam generator, the nodalization (Figure A.1) shows use of only one volume for the entire steam generator secondary side.

   Justify the steam generator modeling and demonstrate that it will produce conservative results. In addition, provide details of qualification of the steam separator model, liquid level model, steam line model, bypass valve sizing, etc. and assess and justify the uncertainty level (or bias) associated with each one of these models. Describe thoroughly the impact of the secondary side modeling on secondary side initiated and dominated transients and justify (through parametric studies) the particular modeling selected.

   b. The topical report stated that inability to match certain primary side parameters when compared to the USAR analyses was attributable to the heat transfer modeling in the SG component. Identify and explain thoroughly the source(s) of these differences and justify not obtaining identical or more conservative results.

3. On a transient-by-transient basis, justify modeling the pressurizer on the unaffected side (the single loop in the model representing three unaffected loops in the plant) instead of the affected side and explain the impact on transient system behavior and conservatism.

4. Justify that the upper head circulation path modeling and predicted flows are realistic.

5. Provide the sequence of events tables for the analyses performed in the Start-Up Test Comparison and the USAR Benchmark Section, comparing the actually measured or current USAR predicted events with WCNOC predicted events indicating the time as well as the key system conditions. Similarly, provide the sequence of events tables for the

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analyses performed in the Enveloping Transient Section.

6. The RETRAN-02 MOD3 SER states that each user is expected to develop and qualify a boron transport model which must be approved. Describe the RETRAN model used for comparison to USAR and envelope main steam line break analyses and provide its qualification analysis.

7. Describe the decay heat model used in comparison to USAR and envelope transient analyses.

2.0 <u>Startup Test Comparisons</u>

Generally the WCGS base model exhibited difficulties in replicating the secondary side behavior recorded during the tests. Therefore, for secondary side initiated transients, the primary side behavior was not well predicted.

2.1 Large Load Reduction

8. Explain thoroughly how this analysis was performed. Discuss the reasons why the RETRAN computed steam dump demand (Figure D.1-1) did not match the test data and the impact of not matching on the results. Explain how this mismatch supports (or fails to support) the RETRAN control system model of the steam dump system.

9. Explain the source(s) of the RETRAN underprediction of the peak in the cold leg temperature at roughly 50 and again at 200 seconds into the test while the coolant average temperature remained overpredicted between 75 to 400 seconds.

2.2 Turbine Trip Without Steam Dump

10a. Explain and justify the reduced SG heat transfer in the RETRAN analysis which resulted in a 55.2°F drop in primary coolant temperature across the steam generator while the test had a 58.7°F delta T.

10b. Check and identify the source of discrepancies between the data on the plots on pg. D-20 and the initial conditions provided in Table D-2 and resubmit any corrected results for review.

2.3 Reactor Coolant Pump Coastdown

11. Explain the necessity for and justify using an initial RC flowrate which was more than 10% less than test data.

3.0 <u>USAR Comparison</u>

12. Explain the need for two WCGS RETRAN models: one for the test comparison and the other for the USAR comparison. Discuss the differences between these two models and their impact on USAR comparison since a simplified model is used for such comparison. Explain which of the benchmark comparison with the test data is

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relevant to the qualification of the evaluation model.

13. Discuss the differences in modeling used in the USAR and RETRAN analysis with emphasis on the following:

    a. reactivity feedback modeling in the two sets of analyses.

    b. Substantiate the statement made in several transient analyses that the differences seen in the computed results are due to "effects of the Doppler coefficient interpolation schemes in combination with the trip reactivity characteristics." What system behavior led the licensee to the conclusion that the difference in RETRAN predicted and USAR analyses was attributable to the difference in the interpolation scheme of Doppler feedback. On a transient-by-transient basis, provide a thorough explanation of how these differences would results in the differences observed between the RETRAN computed and USAR results.

    c. Explain in depth the differences between RETRAN steam generator modeling and those used in the USAR. Discuss further the minimum water volumes required to cover the tubes in the USAR and RETRAN analysis. Provide also initial SG liquid mass and liquid levels assumed in the USAR and RETRAN analyses on a transient-by-transient basis.

    d. With respect to the difference identified on pages B-4 through-6, justify using these models which are not necessarily conservative for some transients.

    e. Discuss the difference in the low-low SG level trip models and setpoints used in analyses. Justify that the RETRAN base model is able to predict the SG mixture level accurately for this use.

3.1 Uncontrolled RCCA Withdrawal at Power

See Q13.b. No other specific questions.

3.2 Complete Loss of Reactor Coolant Flow

See Q13.b.

14. Explain the statement on page B-14 that "the RETRAN pressurizer pressure variation results from...... a more conservative primary-to-secondary heat transfer.."

3.3 Locked Rotor

15. Identify the location of the maximum RCS pressure (Figures B.3-5 and B.3-10) if other than the pressurizer. Explain the source of oscillatory behavior in the RETRAN RCS pressure between 0 - 4 seconds since in the enveloping calculation this behavior was not exhibited.

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Explain the difference of over 20 psia in the initial RETRAN and USAR pressures on Figure B.3-5 as well as more than a 50 psi difference between stated pressure on Table B-3 and the values on Figure B.3-5 (and B-3-10) and discuss their impact on the analysis. Discuss thoroughly the causes for the large underprediction (over 100 psi) of RCS pressure by RETRAN.

3.4 Loss of Load/Turbine Trip

16. Explain the large difference in the pressurizer water volume between the RETRAN and USAR predictions in Figures B.4-3, 8, 13, and 18.

17. Explain, in terms of the SG modeling and primary-to-secondary heat transfer, why the core inlet temperatures are consistently predicted higher in the RETRAN calculations than in the USAR predictions while the coolant average temperature is initially lower in the RETRAN calculations but becomes higher after 30-40 seconds. Explain further the difference in low-low steam generator level trip models and setpoints used in RETRAN and USAR analyses.

3.5 Loss of Normal Feedwater

18. Provide a thorough discussion of this transient by inter-comparing system parameters and identifying the sources of differences in these parameters. Furthermore, provide thorough discussion and justification of the SG heat transfer modeling and discuss it vs. the nominal plant conditions. Compare and justify the initial SG mass and water levels assumed in the USAR and RETRAN analyses.

3.6 Feedwater Line Break

19. Provide information related to the SG secondary side, such as the mass inventory and heat transfer coefficients as a function of tube height. Explain thoroughly the predicted results on the basis of such SG secondary side modeling.

20. Provide further qualification of the pressurizer model (as required by the RETRAN SER) for the situation where the pressurizer goes solid. Justify use of the non-equilibrium pressurizer model for this transient.

21. Provide details of assumption differences between with and without offsite power cases to cause large differences in the faulted cold leg temperature (Figs. B.6-6 and 15) and pressurizer liquid volume (Figs. B.6-3 and 12).

3.7 Main Steamline Break

22. If, as stated, the same moderator temperature coefficients were used, explain the inconsistent trends between the slower cooldown and faster power increase predicted by RETRAN when compared with those by USAR.

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Part 2

4.0 WCNOC Enveloping Transients

1. Provide descriptions of model changes and justify the basis for the following assumptions:

   a. SG tube plugging, the amount of adjustments made to HT areas, RC flow area, fluid volumes, metal masses;

   b. Reduced thermal design flow: the method by which an allowance for future flow degradation in the RCS was determined;

   c. Increased secondary side blowdown: Reference NRC approval of setpoint changes for the main steam safety valves.

2. Provide a table of transient specific actuation setpoints of trips, time delays, trip parameters and initial values.

3. Explain thoroughly what is meant by the statement (p. 7) "the initial reactor power, RCS temperatures, and pressures were adjusted to the maximum allowable value including allowances for calibration and instrument errors consistent with maximizing the challenges to the RCS boundary."

4. Discuss the rational behind setting some initial plant conditions at conservative values and others at "nominal" values for RETRAN analyses. Provide differences between initial conditions used in the RETRAN and DNB analysis.

5. Identify any changes, regardless of magnitude, to transient assumptions and initial conditions (including reactivity coefficients and power profiles) assumed in the enveloping transients from the current USAR analyses on a transient-by-transient basis.

6. Justify the selection criteria for reanalysis of Chapter 15 transients presented in the topical report and the reason why some parametrics were not included (i.e., varying reactivity insertion rates, partial power cases for uncontrolled RCCA Withdrawal at power analysis, analysis at subcritical conditions, break sizes and locations, etc.). If any of these have been already performed, provide detailed results with thorough analysis.

4.1 Uncontrolled RCCA Withdrawal at Power

7. Explain Figure 2.1-8.

4.2 Complete Loss of Reactor Coolant Flow

8. Explain Figure 2.2-2. Explain why the transient was started from 120% flow.

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4.3 Locked Rotor

9. The pressurizer pressure is computed to peak at about 4 seconds into the transient (Fig. 2.3-3) while in the comparison to the current USAR analysis (Section B.3) it was computed to occur prior to 3 seconds with a different pressurization rate. Explain the difference between these two sets of calculations (initial conditions, transient assumptions, etc.).

4.4 Loss of Load/Turbine Trip

10. Which case does the MDNBR plotted on Figure 2.4-6 represent? How do the DNBRs for the other cases differ from this?

11. Explain what reactivity feedback mechanism modeled in Cases 2 and 4 causes the power to increase during the first 5 seconds of the transient. Explain further the reasons why when PZR pressure control is modeled, the PZR pressure increases (Cases 1 and 2) and when the pressure is allowed to increased, the pressure peak is lower by more than 100 psi. Explain why similar inconsistency is predicted in the PZR water volume.

4.5 Loss of Normal Feedwater

12. Discuss the rational used in determining the initial conditions shown on Table V. Justify the value used for the AFW flowrate used.

13. Justify the changes made to this analysis including the reactivity coefficients to cause the system behavior to change from that presented in the comparison with the current USAR prediction.

4.6 Feedwater Line Break

14. Discuss the changes made to this analysis to cause the system behavior to change from those presented in the comparison with the current USAR prediction. Explain Figure 2.6-5.

15. Describe the effect of modeling pressurizer pressure control on the margins to hot leg saturation and pressurizer solid conditions.

4.7 Main Steamline Break

16. Re-analyze this transient using a split-core model and demonstrate that WCNOC's MSLB model is conservative. Discuss, in depth, the mixing and reactivity feedback modeling assumed in the analysis.

ENCLOSURE 5

NRC Interview - January 28, 1992

Core T/H Notes

WCNOC Core T/H Topical - Response to RAI

- History
  - November 1988 - WCNOC Begins Core T/H Methods Development
  - August, 1990 - Core T/H Topical Submitted
  - January, 1992 - RAI Received
  - February 28, 1992 - WCNOC Response to RAI Due

WCNOC Core T/H Topical - Response to RAI

- RAI Questions Fall into 4 Groups
  - Statistical Core Design
  - Modeling Techniques
  - CHF Correlation
  - MAP Limit Methodology

WCNOC Core T/H Topical - Response to RAI

- Statistical Core Design
  - Based on B&W Methodology (BAW 10170-P-A)
  - Response Surface Model Used for Propagation

WCNOC Core T/H Topical - Response to RAI

- Modeling Techniques
  - Options Selected to Yield Limiting T/H Environment
  - Correlation Selection Analysis Dependent
    - Small Heat Transfer Coefficient - Maximum Fuel Temp.
    - Large Heat Transfer Coefficinet - DNB Analysis
  - Model Presented in Topical is not Static
    - Axial Noding May be adjusted to accomadate power distributions.
    - Radial Noding may be adjusted to examine bundle crossflows.

WCNOC Core T/H Topical - Response to RAI

- CHF Correlation
  - WRB-1 correlation presented in Topical
  - Compete analysis of correlation database performed
  - Test section geometries and test conditions obtained directly from vendor

WCNOC Core T/H Topical - Response to RAI

- Maximum Allowable Peaking (MAP) Curves
  - WCNOC MAP Limit Methodology base on B&W Methods
  - MAP generated at several points on Core Safety Limits
    - Intersection of 118% Power and DNB Limits
    - Intersection of SGSV Line and High Pressure DNB Limits
  - Most Restrictive Limits used to Define Final Safety Limit MAP
  - MAP Curves may be Adjusted Downward for Part Power Multiplier

REQUEST FOR ADDITIONAL INFORMATION

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

TOPICAL REPORT TR-90-0025-W01

1. Provide justification for the assumption that the three-step radial power distribution and the modified Core Statepoint-1 used in determining the core-wide protection limits are bounding. What evaluation will be performed to confirm these assumptions for a specific reload cycle? 8 hrs

2. How will it be ensured that the power distribution assumed in the 17-Channel Model (of Section-3) used to determine the DNBR safety limit lines is bounding for a specific reload cycle? 4 hrs

3. How will the cycle-to-cycle variations in fuel design and core loading be accounted for in the VIPRE-01 model? 4 hrs

4. How are assembly rod-wise power distributions which are not octant symmetric, due either to fuel design or global core power distribution, accounted for in the VIPRE-01 model?

5. Does the VIPRE-01 axial representation assume that the MDNBR occurs between the 68 and 130 inch elevations and, if so, how are situations where the MDNBR occurs outside this region treated?

6. Provide the basis for the uncertainties and the assumed (normal and uniform) distributions for the variables given in Table 4-12.

7. How do the Wolf Creek Nuclear Operating Corporation (WCNOC) statistical core design (SCD) and response surface methodology (RSM) differ from the methods described in Reference-16 of the topical report? Explain any differences.

8. In the selection of the random variables from the normal and uniform distributions, are values greater than the 95 percent points selected? If not, how is this simplification accounted for?

9. Provide justification for the use of the K=1.724 95/95 upper tolerance factor for the RSM fitting error. What error is introduced by this assumption?

10. What evaluation will be performed to ensure that the statepoints used in determining the hot-pin protection statistical design limit are bounding for a specific reload cycle?

11. Are the Wolf Creek Generating Station (WCGS) fuel designs to which the WRB-1 correlation will be applied included in the presently approved applications of WRB-1/THINC?

12. Why is the correlation design limit to be used with WRB-1 in VIPRE-01 proprietary? The design limit value is given in WCAP-8567, as well as in the NRC SER, without proprietary brackets, and therefore should not be considered proprietary in this topical report.

13. Why does the number of data points and test series given in WCAP-8762 differ from the number given in Table 2-4? Please justify the use of the smaller number.

14. What tests have been performed to ensure that the M/P data is normal?

15. Is the procedure used to determine the steam generator safety valve (SGSV) line (Equation 5-2) the same as is presently used?

16. Are the WCNOC procedures for determining the MAP curves the same as the BWFC methods? If not, discuss any differences.

17. Is the part-power multiplier used below 75 percent power? If so, provide the basis.

18. Do the three points on the core safety limit lines used to determine the MAP curves provide the most limiting MAPs? For example, since the low pressure MAPs are more restrictive, why wasn't the SGSV limit line MAP calculated on the low pressure DNBR limit line?

19. Why are the curves in Figures 6-13 and 6-15 different?

20. The Chen heat transfer correlation does not result in the highest fuel and clad temperatures in Table 3-59. How will conservative maximum fuel temperatures be calculated in specific transients?

21. Certain combinations of fluid correlations have not been included in the Section-3.3.4 comparisons. How will these cases compare to the base-case thermal-hydraulic model?