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TU ELECTRIC

March 31, 1992

William J. Cahill, Jr.
Group Vice President

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
REQUEST FOR ADDITIONAL INFORMATION ON RXE-91-002
"REACTIVITY ANOMALY EVENTS METHODOLOGY"

REF: 1) Letter from W. J. Cahill, Jr. to the NRC logged TXX-92093 dated, February 14, 1992.

2) Letter from the NRC to Mr. William J. Cahill, Jr. dated January 14, 1992, Requesting Additional Information regarding Topical Report RXE-91-002.

Gentlemen:

Reference 1 transmitted TU Electric's response to 26 of the 28 questions provided in Reference 2. Attached, please find TU Electric's responses to the two remaining questions provided in Reference 2.

Should clarification or additional information regarding responses to the referenced letter be required to enable the Staff to complete its review, contact Mr. Jimmy D. Seawright at 214-812-4375.

Sincerely,

William J. Cahill, Jr.

By: J S Marshall
J. S. Marshall
Generic Licensing Manager

JDS/gj
Attachment

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (2)
Mr. T. A. Bergman, NRR
Mr. B. E. Holian, NRR

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TOPICAL REPORT RXE-91-002

REACTIVITY ANOMALY EVENTS METHODOLOGY

Note: The references, figures, tables, and nomenclature quoted in this response correspond to those provided in Topical Report RXE-91-002.

24. Question

Provide the methodology, predictions and sensitivity studies for the control rod ejection DNBR analyses.

Answer

Note: The tables within the text of this response that are not found within RXE-91-002 are identified by alphabetic character, and are located at the end of the response to this question.

The core T-H model is used to evaluate each CRE event scenario with respect to DNB. A statepoint analysis is performed using boundary conditions for core power, system pressure, core inlet temperature, and core inlet flow rate from the system T-H analysis. The system boundary conditions selected for use in the statepoint analysis correspond to the most limiting event conditions with respect to DNB. A chopped cosine axial power shape is used unless the hot channel axial power shape obtained from the core physics calculations is more limiting.

The DNB analysis results for each case are evaluated against the appropriate base case by utilizing the relationship of the MDNBR to the product of the $F_{\Delta H}$ and the core power. This relationship is unique for a given combination of temperature, pressure, flow rate, and axial power shape, i.e., a given base case. The mathematical relationship is given in Equation 4.3-1, and is restated below for completeness.

$$F_{\Delta H,LIM} = \frac{(P_{DNB}) \cdot (F^N_{\Delta H})}{P_{ST}}$$

where:

P_{ST}	=	Core power at a statepoint
P_{DNB}	=	DNBR limited core power at $F^N_{\Delta H}$, T_{ST}, and PR_{ST}
$F^N_{\Delta H}$	=	Design limit hot channel peaking factor
T_{ST}	=	Core inlet temperature at a statepoint
PR_{ST}	=	RCS pressure at a statepoint
$F_{\Delta H,LIM}$	=	DNBR limited hot channel peaking factor at P_{ST}

For demonstration purposes, the following conditions were used in the core T-H analyses to establish the base cases.

$F^N_{\Delta H}$	=	1.55
T_{ST}	=	565.5°F for HFP 557°F for HZP
PR_{ST}	=	2208 psia
$DNBR_{LIM}$	=	DNBR design limit = 1.35
$\dot{m}_{CORE}$	=	100% Thermal Design Flow for HFP 46% of Thermal Design Flow for HZP

The pressure is held constant throughout the core T-H analysis because the initial value is the most restrictive with respect to DNB. The core inlet temperature remains constant throughout the core T-H analysis because the time to reach MDNBR is much less than the loop transit time. The core inlet flow rate is maintained constant at the minimum expected flow rate for the scenario of interest.

The core T-H analysis is performed using the above stated conditions, in conjunction with a limiting axial power shape, to determine the DNBR limited core power (P_{DNB}) for the core exposure and initial power level of interest, i.e., the base case. The peak core average thermal power (P_{ST}) for a given event scenario is then used to determine the DNBR limited hot channel peaking factor ($F_{\Delta H,LIM}$). The $F_{\Delta H,LIM}$ is used to perform a fuel pin census to determine the number of fuel pins which experience DNB. Any fuel pin having an augmented $F_{\Delta H}$ greater than or equal to $F_{\Delta H,LIM}$ is assumed to experience DNB. The peak core average thermal power from the system T-H analysis and the results of the core T-H analysis/fuel pin census for the cases presented in Section 5.5 are provided in Table A. The corresponding results for the RXE-91-002 Appendix B sensitivity studies are provided in Tables B through D.

TABLE A
Control Rod Ejection DNBR Results

Case	Peak Thermal Power (MW)	Pins in DNB % Total
HFP BOC	3962	0.01
HFP EOC	3902	0.02
HZP BOC	696	0.3
HZP EOC	1013	4.1

TABLE B

HFP Neutronics Parameters Sensitivity Study DNBR Results

Parameter	Variation	BOC HFP Peak Thermal Power MW	BOC HFP Pins in DNB % Total	EOC HFP Peak Thermal Power MW	EOC HFP Pins in DNB % Total
Nominal		4032	0.01	4015	0.02
DTC	-10%	4057	0.01	4032	0.02
	+10%	4009	0.01	4001	0.02
MTC, pcm/°F	+3	4058	0.01	4029	0.02
	-3	4008	0.01	4004	0.02
ℓ^*, μsec	+5	4032	0.01	4017	0.02
	-5	4031	0.01	4014	0.02
β_{eff}	-5%	4047	0.01	4016	0.02
	+5%	4017	0.01	4015	0.02
Ejected Control Rod Worth	+10%	4089	0.01	4075	0.16
	-10%	3976	0.01	3961	0.02
Ejection Time	Doubled	4030	0.01	4013	0.02
	Halved	4032	0.01	4016	0.02
Reactor Trip Delay Time, seconds	+0.5	4078	0.01	4015	0.02
	-0.5	3962	0.01	4016	0.02
Scram Worth	+10%	4030	0.01	4015	0.02
	-10%	4034	0.01	4015	0.02
Time Step Size, seconds	0.01	4035	0.01	4018	0.02
Reload β's & λ's		4039	0.01	4020	0.02

TABLE C

HZP Neutronics Parameters Sensitivity Study DNBR Results

Parameter	Variation	BOC HZP Peak Thermal Power MW	BOC HZP Pins in DNB % Total	EOC HZP Peak Thermal Power MW	EOC HZP Pins in DNB % Total
Nominal		680	0.1	1009	4.1
DTC	-10%	765	1.3	1138	7.2
	+10%	612	0.1	907	2.9
MTC, pcm/°F	+3	732	0.6	1010	5.7
	-3	637	0.1	1003	4.1
ℓ^*, μsec	+5	680	0.1	999	4.1
	-5	681	0.1	1030	5.7
β_{eff}	-5%	715	0.4	1047	5.7
	+5%	644	0.1	971	4.1
Ejected Control Rod Worth	+10%	813	2.1	1203	7.2
	-10%	539	0.1	821	0.8
Ejection Time	Doubled	680	0.1	1009	4.1
	Halved	680	0.1	1009	4.1
Reactor Trip Delay Time, seconds	+0.5	700	0.3	1009	4.1
	-0.5	655	0.1	1009	4.1
Scram Worth (-5% for EOC HZP)	+10%	678	0.1	1009	4.1
	-10%	682	0.2	1009	4.1
Time Step Size, seconds	0.01	698	0.3	692	0.1
Reload β's & λ's		684	0.2	1010	4.1

TABLE D

Thermal-Hydraulic Parameters Sensitivity Study DNBR Results

Parameter	Variation	BOC HFP Peak Thermal Power MW	BOC HFP Pins in DNB % Total	EOC HFP Peak Thermal Power MW	EOC HFP Pins in DNB % Total
Nominal		4032	0.01	4015	0.02
Fuel Temperature[1]	+50%	4081	0.01	4058	0.10
	-50%	3931	0.01	3955	0.02
Fuel Pellet, # mesh pts.	10	4028	0.01	4011	0.02
Inlet Temperature, °F	+5.5	4032	0.01	4017	0.02
	-5.5	4031	0.01	4014	0.02
RCS Pressure, psi	+30	4032	0.01	N/A	N/A
	-30	4032	0.01	N/A	N/A

Parameter	Variation	BOC HZP Peak Thermal Power MW	BOC HZP Pins in DNB % Total	EOC HZP Peak Thermal Power MW	EOC HZP Pins in DNB % Total
Nominal		680	0.1	1009	4.1
Fuel Temperature[1]	+50%	753	9.9	1234	7.2
	-50%	533	0.1	668	0.1
Fuel Pellet, # mesh pts.	10	678	0.1	976	4.1
Inlet Temperature, °F	+5.5	682	0.2	1013	5.7
	-5.5	678	0.1	1005	4.1
RCS Pressure, psi	+30	680	0.1	1009	4.1
	-30	680	0.1	1009	4.1

[1] Fuel rod gap conductance used for variance

25. <u>Question</u>

In the rod ejection accident analysis, the use of a film boiling heat transfer correlation is conservative for fuel enthalpy calculations, but is nonconservative for heat flux predictions in DNBR analyses. How is the heat transfer calculation performed in the DNBR analysis?

<u>Answer</u>

The hot spot model uses a film boiling heat transfer correlation for the fuel enthalpy calculations. The hot spot model is not used for the DNB analyses. The heat transfer calculation for the DNB analysis is based on the core average thermal power response of the system T-H analysis. The system T-H model uses a subcooled nucleate boiling correlation to predict the fuel pin heat flux.