Log # TXX-92093
File # 10010
915

TU ELECTRIC

February 14, 1992

William J. Cahill, Jr.
Group Vice President

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 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: Letter from the NRC to Mr. William J. Cahill, Jr. dated January 14, 1992, Requesting Additional Information regarding Topical Report RXE-91-002

Gentlemen:

Attached, please find TU Electric's responses to 26 of the 28 questions provided in the referenced letter. The responses to the remaining two questions require additional analyses and therefore require additional time for completion. TU Electric will provide the responses to those questions by March 31, 1992.

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: DRWoodlan

D. R. Woodlan
Docket Licensing Manager

JDS/grp
Attachment

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

D029
1/1

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.

1. <u>Question</u>

Have the TUE methods and correlations of References 3-7 been approved for the applications required by the reactivity anomaly methodology of RXE-91-002?

<u>Answer</u>

References 3 through 7 of RXE-91-002 are currently under review by the NRC Staff, and have not yet been approved for the applications required by RXE-91-002. The intent of RXE-91-002 is to present the methodology unique to the analysis of the reactivity anomaly events and is therefore independent of the analytical methodology presented in each of the referenced documents. Regardless of the approval status of these documents, NRC-approved methodology will be used to develop the inputs necessary to perform the analyses presented in RXE-91-002. Thus, the approval of the analytical methodologies presented in the referenced reports should not be a prerequisite to the acceptance of the methodology presented in RXE-91-002.

2. Question

Discuss the validation of the hot-spot calculation under the fuel melt conditions that occur in the rod ejection accident. In this case, how is the fission gas release determined from the fraction of fuel melt?

Answer

TU Electric utilizes NRC-approved material properties and heat transfer mechanisms for the hot-spot calculations. The application of the fuel material properties in conjunction with the use of RETRAN-02 constitutes sufficient validation of the hot-spot model under the conditions of fuel melt.

The fuel material properties are extracted from the information provided in Reference 24. The specific fuel material properties used in the hot-spot model include the enthalpy, specific heat capacity, thermal conductivity, and melting temperature. Included within the development of these fuel material properties are the changes that occur as a result of fuel melt and the effects of fuel exposure on the fuel melting temperature.

The conduction and convection heat transfer models within RETRAN-02 have been approved by the NRC for use as stated in References 21 and 22. These heat transfer models in conjunction with the fuel material properties and a conservative fuel pellet power generation profile are used to calculate the energy deposition and heat transfer for each of the ten concentric fuel regions [see Figure 3.7-1]. When the fuel temperature within a region satisfies the temperature criterion for fuel melt, the amount of energy deposited in the fuel region and the amount of energy transferred to the next fuel region are calculated based on the phase change characteristics specified by the input material properties.

The fission gas release from the melted fuel is used to derive a portion of the source term needed to determine the offsite radiological dose. The specifications in Appendix B of Regulatory Guide 1.77 [Reference 14], with the limitations specified in the CPSES-1 FSAR, are used to determine the extent and type of fission gas release from the melted fuel. For conservatism, any fuel region that attains or exceeds the fuel melting temperature is assumed to be fully melted. Additional conservatism is included in the offsite radiological dose calculation by assuming that 50% of the iodine contained in the melted fuel is available for release from the plant secondary.

3. Question

The fuel rod gap conductance and radial power distribution affect the transient moderator and Doppler feedbacks and the margin to fuel enthalpy and DNBR limits. How will these parameters be determined to insure conservative CPSES-1 licensing analyses?

Answer

The fuel rod gap conductance used for the system thermal-hydraulic analysis of the reactivity anomaly events is set in accordance with the results of various sensitivity cases. These sensitivity cases are used to establish the direction for conservatism, i.e., minimum or maximum fuel rod gap conductance, for each of the specific event analyses.

The core radial power distribution is not explicitly modelled as part of the point kinetics solution to the core power response for the reactivity anomaly event analyses. Instead, the feedback resulting from the point kinetics solution to the problem is conservatively modelled by selecting limiting Doppler and moderator reactivity coefficients or defects for the event analyses. The methodology described in Section 4.2 of RXE-91-002 is used to calculate the Doppler and moderator reactivity coefficients and defects.

For some events, such as the control rod ejection event, it is often desirable to more accurately model the Doppler reactivity feedback while still maintaining an overall conservative feedback response. The core power distribution is an important contributor to the overall transient reactivity feedback due to the non-uniform effect of the ejected control rod on the reactor core power. A Doppler Weighting Factor (DWF) is employed in these instances to correct for the increased Doppler feedback associated with the spatial effects of a non-uniform fuel temperature rise. The calculation of a DWF is performed in accordance with the methodology described in Section 4.8.3 of RXE-91-002.

4. <u>Question</u>

What uncertainty allowance will be included in the temperature feedback coefficients, control rod worths, boron worth, and power distribution peaking factor input to the CPSES-1 reactivity anomaly licensing analyses?

Answer

The appropriate limiting values for the Moderator Temperature Coefficient (MTC) protected by the Technical Specifications and the Core Operating Limits Report are used in all reactivity anomaly analyses, except for the control rod drop event analysis. The control rod drop event analysis uses the calculated value of MTC with an uncertainty no less than that approved in Reference 1. The uncertainty applied to the Doppler temperature feedback and to the boron worth is 10% as approved in Reference 1.

The uncertainty applied to the calculated control rod worth is determined on a cycle-specific and an event-specific basis. The differential control rod worth and the ejected control rod worth use an uncertainty of 15% to increase calculated control rod worth. The trip reactivity, including the effect of a stuck control rod, is decreased by 10% from the calculated nominal value. For the control rod drop event analysis, the inserted control bank worth is conservatively calculated with 1) the control banks at their full power insertion limit, and 2) a power distribution corresponding to an axial offset at the upper end of the normal operating axial offset bands. No uncertainty is applied to the worth of the dropped control rod because the dropped control rod worth is the independent variable used to parameterize the post-drop to pre-drop $F_{\Delta H}$ ratio.

The augmentation factors described in Sections 4.2 and 4.8.3 of RXE-91-002 are applied to the power distribution peaking factors. Since this analytical approach implicitly assumes the power distribution peaking factors are at the licensed limit for the time of maximum peak during normal operation, no additional uncertainty is required. The input to the control rod drop event analysis is even more conservative in that it effectively assumes that the core is at the $F_{\Delta H}$ limit at each cycle exposure.

5. Question

How do the TUE reactivity event analyses of RXE-91-002 differ from the CPSES-1 FSAR analyses with respect to initial/boundary conditions and system performance?

Answer

The initial/boundary conditions used in the TU Electric reactivity anomaly event analyses are essentially the same as the CPSES-1 FSAR analyses. The system performance of the major RCS parameters, i.e., core power, RCS pressure, and core average fluid temperature, are also analogous to the CPSES-1 FSAR analyses. Although the general trend of each event is similar to that of the corresponding CPSES-1 FSAR analysis, a direct numerical comparison of the analyses is not appropriate. The reactivity anomaly event analyses presented in RXE-91-002 employ the analytical methodologies developed by TU Electric, while the CPSES-1 FSAR analyses utilize methodology developed by Westinghouse. The TU Electric and Westinghouse reactivity anomaly event methodologies have many similarities. However, a few significant methodology differences do exist, as noted below.

1. The DNBR results presented in the CPSES-1 FSAR utilize the Westinghouse W-3R correlation while the TU Electric DNBR results utilize the TUE-1 correlation. A one-to-one comparison of the DNBR results is therefore not meaningful. Instead, a more meaningful comparison is made by stating that those events in which DNB is

precluded in order to satisfy the acceptance criterion, the MDNBR remains greater than the applicable correlation limit. For those events in which the specified correlation limit is violated, a conservative estimate of the number of failed fuel pins is used to determine the resulting offsite radiological consequences. These offsite radiological consequences are then shown to be within the acceptance criterion specified for the event of interest.

2. The development of N-16 protection system trip setpoints used in the TU Electric event analyses use the TUE-1 DNB correlation [References 5 and 6] in conjunction with the TU Electric N-16 setpoint methodology [Reference 3]. The CPSES-1 FSAR event analyses use the Westinghouse W-3R correlation in conjunction with the Westinghouse N-16 setpoint methodology to derive the setpoints. A different trip setpoint affects the event results by changing the time (and hence the system conditions) at which the reactor trip occurs and, potentially, by changing the trip function providing reactor protection. A more meaningful comparison is achieved by stating that the results of each analysis are within the acceptance criterion specified for the event of interest.

3. The reactivity anomaly event analyses performed by TU Electric use a point kinetics model to predict the reactor core power response. Several of the event analyses (e.g., control rod ejection and single control rod withdrawal) presented in the CPSES-1 FSAR use a one-dimensional kinetics model to determine the core power response. The development of input parameters for each kinetics model is sufficiently different that

small differences in the predicted event response may exist due to the application of specific conservatism.

6. <u>Question</u>

Justify the assumption that the maximum (full power) $F_{\Delta H}$ - statepoint provides the most limiting DNBR statepoint for the misaligned control rod analysis. For example, do statepoints having higher rod worths and/or maximum excess reactivity provide a closer approach to the DNBR limit?

<u>Answer</u>

Because the misaligned control rod event is a static event, the only parameter of importance is the power distribution resulting from the misalignment. This power distribution is significantly influenced by the power peaking of the unperturbed case. Excess reactivity at a statepoint influences the absolute value of the peak power, primarily through moderator reactivity feedback. However, the increase in peaking as a result of the misaligned control rod is relatively insensitive to moderator reactivity feedback. Hence, the effects of excess reactivity are included in the determination of the reference statepoint based on the maximum full power $F_{\Delta H}$ statepoint.

The combination of a high control rod worth and a high $F_{\Delta H}$ may result in a more limiting analysis, especially when assuming the misaligned control rod to be withdrawn. In addition to the statepoint with the overall maximum full power $F_{\Delta H}$, the analysis must evaluate each full power statepoint with a local maximum $F_{\Delta H}$ and the statepoint with the maximum inserted control bank worth to ensure the identification of the limiting statepoint.

7. Question

Describe the screening calculations used to identify the most limiting misaligned control rod and fuel assembly replacement. How is the uncertainty in these calculations accommodated in the TUE methodology?

Answer

The most limiting misaligned control rod is identified by analyzing each control rod misalignment with a two-dimensional nodal model using SIMULATE-3 [Reference 16]. These calculations are performed for rodded and unrodded configurations allowed at operating conditions. The pin power distributions of each of the resulting radial slices are combined using an appropriate power sharing for each slice to provide an estimate of the radial peaking factor for the misaligned case. Each of the potentially limiting misaligned control rods is then evaluated with a three-dimensional model using SIMULATE-3, thereby alleviating the need for a "screening" uncertainty.

For the misloaded fuel assembly event, each misloaded fuel assembly considered is analyzed with a two-dimensional nodal model using SIMULATE-3. The resulting assembly relative power distribution is compared to an assembly relative power distribution generated for the correctly loaded core. The resulting assembly-wise differences are evaluated, using the criteria outlined in the response to Question 8, to determine which misloadings would be detected. Non-detectable misloaded fuel assemblies are evaluated, using the two-dimensional model, to determine the consequences of full power operation with the misloaded assembly. The uncertainty associated with the screening calculations is accommodated by evaluating, in detail, all potential misloadings which are calculated to approach the event-specific acceptance criteria using the three-dimensional nodal model.

8. <u>Question</u>

What criteria are used to determine if a fuel misloading would be detected? Describe how instrument uncertainty, power tilts and failed detectors are accounted for.

<u>Answer</u>

The relative power distribution for the misloaded fuel assembly scenario of interest is compared to the relative power distribution for the correctly loaded core design. This comparison determines the predicted difference for instrumented fuel assemblies. Any scenario which results in any instrumented fuel assembly exhibiting a predicted difference in relative power greater than the acceptance criteria used to evaluate a flux map is considered to be detected.

The instrument uncertainty is small in comparison to the flux map acceptance criteria. Therefore, the analysis does not consider an additional penalty for the instrument uncertainty.

Although quadrant power tilts are another means of detecting a misloaded assembly, the analysis does not credit them. Any design asymmetries are explicitly modeled for the full core calculations and are therefore reflected in both the calculated nominal and perturbed power distributions.

The analysis assumes no failed detectors because the initial flux map is obtained immediately after completion of the refuelling outage. All necessary maintenance on the flux mapping system would have been performed at this time.

9. Question

How is the effect of the fuel burnup dependence of the assembly reactivity accounted for in the selection of the limiting fuel assembly misloading? Are cycle depletions performed for all potential misloadings?

Answer

The effects associated with fuel burnup are considered for non-detectable misloaded fuel assembly scenarios in which one of the assemblies contains burnable absorbers. Those scenarios are screened by performing a cycle depletion calculation using the two-dimensional model. Depletions past equilibrium xenon conditions are not required for other potential misloadings.

10. Question

How is a limiting power distribution determined for the misaligned control rod and misloaded fuel assembly analyses? Are the neighboring assemblies initially operating at the DNBR limit?

Answer

The relative power of each fuel pin is calculated with the three-dimensional nodal model. These calculated relative pin power distributions are then increased with the augmentation factor discussed in Section 4.2 of RXE-91-002. The limiting power distribution is the distribution resulting in the maximum relative pin powers.

Neighboring fuel assemblies are not assumed to be initially operating at the DNBR limit.

11. Question

In the calculation of the physics parameters for the hot-zero-power control rod withdrawal analysis, in what sense are the xenon-free beginning-of-cycle and equilibrium-xenon end-of-cycle conditions bounding?

Answer

The xenon-free beginning-of-cycle and equilibrium-xenon end-of-cycle conditions are not intended to be bounding. Instead, these conditions are selected to reflect the conditions prior to startup. The conservatism present in the core physics calculations comes from not crediting the Doppler reactivity feedback effects resulting from the fuel temperature increase prior to the predicted time of minimum DNBR. In addition, the differential control rod worth and core peaking factors calculated assuming a control rod bank overlap of 100% are significantly greater than the corresponding values calculated using the nominal overlap.

12. Question

How is the limiting rod determined for the single rod withdrawal analysis? At what power is the single rod withdrawal event analyzed?

Answer

The limiting control rod for the single control rod withdrawal event analysis is selected based on calculations similar to those used to determine the limiting control rod for the misaligned control rod event analysis (see the response to Question 8). The basis for the selection differs in that the limiting control rod for the single control rod withdrawal event is the control rod resulting in the maximum number of fuel pins experiencing DNB. This control rod may not correspond to the control rod resulting in the greatest $F_{\Delta H}$, as used for the misaligned control rod event analysis.

The power level used as input to the DNBR analysis for the single control rod withdrawal event is determined from an interpolation/extrapolation of the control rod bank withdrawal at power event results. Because the TU Electric control rod withdrawal event methodology uses a point kinetics solution to model the core average power response, the predicted power response resulting from the withdrawal of a single control rod will be identical to that resulting from the withdrawal of an entire control rod bank at an equivalent reactivity insertion rate. The control rod withdrawal at power event is analyzed for a matrix of event scenarios that include several initial power levels, a wide range of reactivity insertion rates, and a variety of reactivity feedback combinations. The single control rod withdrawal event analysis utilizes these event results to determine the time at which the peak core average power occurs as a function of initial power and reactivity insertion rate. The core inlet temperature, RCS pressure, and core average power level existing at the time of peak core average power are used, in conjunction with the event-specific peaking factors, as input parameters to the DNB analysis.

13. <u>Question</u>

Provide additional detail and qualification for the discrete ordinates method used to determine the excore response in the dropped rod analysis.

<u>Answer</u>

Note: The tables and references 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 discrete ordinates calculations use the GIP, ANISN, and DOT 4.3 [References A, B, and C, respectively] computer codes with the ELXSIR cross sections [Reference D]. The calculations are performed in two steps: 1) qualification of methods by performance of the Pool Critical Assembly Problem (PCA) [References E and F] and 2) calculation of the total flux at the excore detector location for CPSES-1 Cycle 1.

Data describing the PCA are given in Reference E. These data include absolute source spectra and material and geometry descriptions. References E and F contain measured results for various nuclear reactions used in pressure vessel dosimetry. Three-dimensional results are synthesized using DOT XY, DOT XZ, and ANISN Z calculations. The methodology for synthesization is described in References G and H. Tables A and B compare the results of the PCA calculations performed by TU Electric and the measured data from Reference E.

The CPSES-1 Cycle 1 excore detector response calculations used the same methodology as the PCA calculations. The CPSES DOT calculation employed R,θ geometry to accurately model excore materials such as the pressure vessel. DOT adjoint calculations were performed to determine a total flux response function at the excore detector location. Determination of the total flux required transforming the X-Y pin by pin relative power distributions from SIMULATE-3 [Reference 16] into the CPSES DOT model using the DOTSOR code [Reference I] and folding the source function from DOTSOR with the adjoint response using the TIMEPATCH code [Reference J].

Excore detector tilts for the dropped control rod conditions were determined as the ratio of the perturbed total flux (rodded condition) to the unperturbed total flux (unrodded condition). Calculations were performed for 22 configurations (11 BOC and 11 EOC). Excore detector tilts determined in this manner were used to confirm the algorithm described in Section 4.7.3 of RXE-91-002. The average difference between the discrete ordinates results and the algorithm results for the 22 cases is 1.3 percent.

Table A

PCA Results
Comparison of Calculated and Measured Nuclear Reaction Rates: Np-237(n,f), Flux > 1 MeV, Flux > 0.1 MeV, and DPA[1].

NP-237(n,f)

	MEASURED	CALCULATED	C/M[2]
TSF[3]	8.371E-06	7.05995E-06	0.84343
PVF	2.939E-07	2.48906E-07	0.84694
1/4 T	1.174E-07	1.13199E-07	0.96402
1/2 T	6.547E-08	6.18091E-08	0.94410
3/4 T	3.411E-08	3.33432E-08	0.97746

Flux > 1 MeV

	MEASURED	CALCULATED	C/M
TSF	3.71000E-06	3.37922E-06	0.91084
PVF	1.33000E-07	1.16442E-07	0.87550
1/4 T	4.30000E-08	4.37263E-08	1.01689
1/2 T	2.07000E-08	2.04506E-08	0.98795
3/4 T	9.11000E-09	9.32052E-09	1.02311

FLUX > 0.1 MEV

	MEASURED	CALCULATED	C/M
TSF	6.940E-06	5.99870E-06	0.86437
PVF	2.490E-07	2.09808E-07	0.84260
1/4 T	1.390E-07	1.31693E-07	0.94743
1/2 T	9.350E-08	8.68525E-08	0.92890
3/4 T	5.570E-08	5.55452E-08	0.99722

DPA (BARNS)

	MEASURED	CALCULATED	C/M
PVF	1.940E-04	1.68955E-04	0.87090
1/4 T	7.510E-05	7.00332E-05	0.93253
1/2 T	4.270E-05	3.86471E-05	0.90508
3/4 T	2.260E-05	2.16791E-05	0.95925

---

1 DPA, displacements per atom.
2 C/M, Calculated/Measured.
3 Column 1 gives the experimental measurement locations:

TSF	outer face of the thermal shield:
PVF	inner face of the pressure vessel simulator;
1/4 T	one fourth depth of the pressure vessel simulator;
1/2 T	one half depth of the pressure vessel simulator;
3/4 T	three fourths depth of the pressure vessel simulator.

Table B

PCA Results
Comparison of Calculated and Measured Reaction Rates:
Ni-58(n,p), Al-27(n,a), In-115(n,np), and U-238(n,f)

NI-58 (n,p)

	MEASURED	CALCULATED	C/M[1]
TSF[2]	6.03860E-07	5.25404E-07	0.87008
PVF	2.39800E-08	2.00522E-08	0.83620
1/4 T	5.50450E-09	5.02819E-09	0.91347
1/2 T	2.18000E-09	1.96897E-09	0.90320
3/4 T	7.74990E-10	7.65009E-10	0.98712

AL-27 (n,alpha)

	MEASURED	CALCULATED	C/M
TSF	5.40310E-09	3.98931E-09	0.73834
PVF	3.08850E-10	2.23617E-10	0.72403
1/4 T	7.06450E-11	5.67332E-11	0.80307
1/2 T	2.84000E-11	2.31291E-11	0.81441
3/4 T	1.05790E-11	9.17348E-12	0.86714

IN-115(n,np)

	MEASURED	CALCULATED	C/M
TSF	1.013E-06	8.86838E-07	0.87542
PVF	3.629E-08	3.07148E-08	0.84642
1/4 T	1.072E-08	1.03376E-08	0.96466
1/2 T	4.971E-09	4.66913E-09	0.93933
3/4 T	2.155E-09	2.08667E-09	0.96847

U-238 (n,f)

	MEASURED	CALCULATED	C/M
1/4 T	1.864E-08	1.63920E-08	0.87961
1/2 T	8.204E-09	7.00725E-09	0.85407
3/4 T	3.385E-09	2.95540E-09	0.87296

---

[1] C/M, Calculated/Measured.
[2] Column 1 gives the experimental measurement locations:

TSF	outer face of the thermal shield:
PVF	inner face of the pressure vessel simulator;
1/4 T	one fourth depth of the pressure vessel simulator;
1/2 T	one half depth of the pressure vessel simulator;
3/4 T	three fourths depth of the pressure vessel simulator.

References

A. L. A. Hassler and N. M. Hassan, "GIP-Group Organized Cross Section Input Program Manual," Babcock & Wilcox Report No. NPGD-TM-456, Rev. 6, March 1985.

B. L. A. Hassler and N. M. Hassan, "ANISNBW - A One dimensional Discrete Ordinates Transport Code," Babcock & Wilcox Report No. NPGD-TM-491, Rev. 6, September 1987.

C. L. A. Hassler and N. M. Hassan, "DOT43-Two Dimensional Discrete Ordinates Transport Code (B&W Version of RSIC/ORNL Code DOT 4.3),' Babcock & Wilcox Report No. NPD-TM-24, Rev. 2, October 1987.

D. R. E. Maerker, et al., "The ELXSIR Cross Section Library for LWR Pressure Vessel Irradiation Studies: Part of the LEPRICON Computer Code System," EPRI Report No. EPRI NP-3654, September 1984.

E. W. N. McElroy, ed., "LWR Pressure Vessel Surveillance Dosimetry Improvement Program: PCA Experiments and Blind Test," NUREG/CR-1861 (HEDL-TME-80-87), July 1981.

F. F. W. Stallman, et al., "Reactor Calculation Benchmark PCA Blind Test Results," NUREG/CR-1872 (ORNL/NUREG/TM-428), January, 1981.

G. R. E. Maerker, M. L. Williams, and B. L. Broadhead, "Accounting for Changing Source Distributions in Light Water Reactor Surveillance Dosimetry Analysis," Nuc. Sci. Eng., 94, 291, 1986.

H. R. E. Maerker, et al., "Application of the LEPRICON Methodology to the Arkansas Nuclear One, Unit 1 Reactor," EPRI Report No. EPRI NP-4469, February 1986.

I. M. L. Williams, "DOTSOR: A Module in the LEPRICON Computer Code System for Representing the Neutron Source distribution in LWR Cores," EPRI Research Project 1399-1, Interim Report, December 1985.

J. R. E. Maerker, M. L. Williams, B. L. Broadhead, "TIMEPATCH : A Module in the LEPRICON Computer Code System for Evaluating Effects of Time-Dependent Source Distributions in PWR Surveillance Dosimetry," EPRI Research Project 1399-1, Interim Report, December 1985.

14. Question

Why isn't the Doppler coefficient included as core physics input in Figure 4.7-1? Is a conservative Doppler coefficient used?

Answer

Figure 4.7-1 illustrates the TU Electric control rod drop analytical methodology in terms of the parameters varied for each scenario analyzed. As such, the variables listed under the heading of "Core Physics Parameters" coincide with the parameters used to characterize a specific control rod drop scenario. The same conservative Doppler coefficient is assumed for each control rod drop scenario; therefore, the Doppler coefficient is not listed among these variables.

15. Question

How is the limiting rod determined in the dropped rod analysis, and how are multi-rod drops treated?

Answer

Both single and multiple control rod drops are considered when determining the $F_{\Delta H,ST}$ and the excore detector tilt for each control rod drop scenario. The control rod drop event analysis utilizes a conservative curve of excore detector tilt as a function of dropped control rod worth. This curve bounds all excore detector tilts resulting from a control rod drop event, including those from multiple control rod drops. The use of a bounding excore detector tilt function not only removes the location and exposure dependency from the calculation of the excore detector tilt, but also removes the dependence on the number of control rods dropped.

The excore detector tilt function is coupled with a model of the Nuclear Instrumentation System to produce a conservative excore detector response during the system thermal-hydraulic analysis. Consequently, the generation of the generic statepoints conservatively accounts for the number of control rods involved in a specific control rod drop scenario.

All unique, plausible control rod drop combinations are considered in the event analysis, including drops of one, two, three, and four control rods from each control rod group. Each control rod drop combination is evaluated to ensure the DNBR acceptance criterion is met. The DNBR acceptance criterion is satisfied for the event by demonstrating that $F_{\Delta H,ST}$ is less than $F_{\Delta H,LIM}$ for each specific event scenario. The event scenario exhibiting the minimum margin between $F_{\Delta H,LIM}$ and $F_{\Delta H,ST}$ is characterized as the limiting case.

16. Question

In the control rod drop analysis, how is the additional uncertainty due to the error introduced in the matrix-method, by interpolating on the input dropped rod worth, inserted bank worth and the moderator temperature coefficient, accounted for?

Answer

The interpolation using the matrix of generic statepoint parameters is performed as part of a screening process to determine the most limiting control rod drop scenario. The physics parameters for this scenario, as identified by the screening process, are then used as input to a system T-H analysis to determine the exact set of corresponding statepoints. These specific statepoints are then used as input to a core T-H analysis to determine the $F_{\Delta H,LIM}$ for the scenario, and the corresponding margin with respect to $F_{\Delta H,ST}$. Any error introduced by the interpolation scheme is removed by performing the specific statepoint analysis. Therefore, the additional uncertainty introduced by interpolation is not included as part of the analysis methodology.

17. Question

In the control rod drop analysis, the SCU methodology considers the uncertainties in the core power, inlet temperature and pressure to be independent and combines these uncertainties statistically. In fact, these calculated variables and their uncertainties are coupled through the neutronic/thermal-hydraulic dynamics of the control rod drop transient and are not independent. The calculation of the MDNBR uncertainty factor using the Appendix A SCU method should employ variables whose uncertainties are independent. The application of the SCU method should include the uncertainties used and their bases.

Answer

The responses of the core power, temperature, and pressure are coupled through the neutronic/thermal-hydraulic dynamics of the system. However, the uncertainty of each variable due to steady-state fluctuations, measurement uncertainties, etc., is independent of the thermodynamic state of the system. Furthermore, these uncertainties are well quantified with well-defined distributions. Therefore, the Statistical Combination of Uncertainties (SCU) methodology as described in Appendix A of RXE-91-002, does employ uncertainties which are independent of the transient response.

The application of the SCU method in a licensing submittal will include a description of the uncertainties used and their bases. The calculations in RXE-91-002 use the current CPSES-1 licensing basis uncertainties for the process parameters to demonstrate the use of the SCU methods.

18. Question

Since DNBR is a required prediction for the rod ejection transient, why aren't the RCS temperature and pressure selected conservatively?

Answer

The RCS temperature and pressure selected for use in the core thermal-hydraulic (i.e., DNB) analysis of the control rod ejection event are selected in a conservative manner. As stated on page 4-47 of RXE-91-002, "the limiting system T-H analysis conditions for core inlet temperature and RCS pressure" are used as inputs to the DNB analysis. This statement is not to be interpreted as a contradiction to the statements provided on page 4-40 of RXE-91-002, regarding the initial RCS fluid conditions, but rather as a clarification of the statements.

19. Question

The treatment of spatial effects in reduced-dimension analyses of the control rod ejection accident is extremely complex and must be validated by comparison to spatial kinetics calculations. Demonstrate that the TUE CPSES-1 point kinetics analysis of the rod ejection transient is conservative relative to a detailed spatial kinetics solution.

Answer

The treatment of spatial effects in reduced-dimension analyses of the control rod ejection accident is indeed an extremely complex issue. The validation of the reduced-dimension control rod ejection event analyses by comparison to spatial kinetics calculations has been documented in References 26 and 27. The conservatism inherent to the reduced-dimension analysis is the direct result of the derivation and/or selection of the input parameters. For the control rod ejection event, the main contributors to the conservatism of the analytical results are:

1. The calculation of the ejected rod worth;
2. The calculation of the Doppler reactivity feedback;
3. The calculation of the core peaking factors; and,
4. The selection of bounding values, as a function of core exposure, to represent other input parameters.

Each of these parameters is developed in accordance with the methodology described in RXE-91-002.

The calculation of the ejected rod worth entails the use of multi-dimension core physics analyses in conjunction with conservative analytical assumptions. These assumptions include the use of adverse axial power distributions (at HFP) to increase the worth of the ejected control rod, the consideration of potential control rod misalignments to increase the worth of the ejected control rod, and the use of an augmentation factor of 15% to further increase the calculated ejected control rod worth.

The Doppler reactivity feedback model used in the analysis of the control rod ejection event is separated into two distinct parts. TU Electric utilizes multi-dimension core physics analyses to compute each portion of the Doppler reactivity feedback. The first part of the model involves calculating the core average Doppler reactivity feedback. This calculation is performed in accordance with the methodology described in Section 4.2 of RXE-91-002. The system T-H analysis conservatively assumes a minimum Doppler reactivity defect, as a function of core exposure, from HZP to HFP.

The second portion of the Doppler reactivity feedback model involves calculating a Doppler Weighting Factor (DWF). The DWF is calculated in accordance with the methodology described in Section 4.8.3 of RXE-91-002. The conservatism inherent to the use of the calculated DWF derives from the fact that the DWF is calculated based on the ejected control rod worth prior to augmentation.

The calculation and use of the core peaking factors is the most important contributor to the overall conservatism of the reduced-dimension analysis. The calculation of the total core peaking factor, F_Q, includes two forms of conservatism. The first form of conservatism results from the use of steady-state thermal-hydraulic feedback in the calculation instead of a transient thermal-hydraulic feedback. This approach derives no benefit from the thermal-hydraulic feedback resulting from the redistribution of power during a control rod ejection event that subsequently reduces the calculated transient F_Q. The second form of conservatism involves the application of a very conservative augmentation factor to increase the calculated F_Q value. The method used to augment the calculated F_Q is described in Section 4.8.3 of RXE-91-002.

In addition to the conservatism present in the calculation of the F_Q, the application of this value to the system T-H analysis and the hot-spot analysis is also performed in a conservative manner. The application of the calculated F_Q to the system T-H analysis is related to the use of the DWF. Because the DWF corrects for the increased Doppler reactivity feedback associated with the spatial effects of a non-uniform fuel temperature rise, an increase to the calculated F_Q results in a corresponding increase to the DWF. An increase to the DWF subsequently results in a lower predicted value for the peak power and a lower power history curve, i.e., a lower value for the integrated full power seconds (FPS). Therefore, the calculated F_Q prior to augmentation is used to calculate the smallest DWF for the specific scenario of interest. The augmented F_Q is then used in the hot spot analysis to determine the extent of fuel melt.

The application of the calculated peak F_Q value to the hot-spot analysis is also conservative with respect to the multi-dimension analysis. The TU Electric methodology for the hot-spot analysis includes the assumption that the pre-ejected peak F_Q and the post-ejected peak F_Q occur at exactly the same core location. This assumption is conservative because multi-dimension analysis of the redistribution of power resulting from the ejection of a control rod, finds the location of the post-ejected peak F_Q

to be different from the location of the pre-ejected peak F_Q. The assumption that the pre-ejected peak F_Q and the post-ejected peak F_Q are situated at exactly the same core location guarantees that the hot-spot analysis is performed at the conditions of maximum initial fuel temperature and energy deposition. Thus, the predicted fuel temperature and enthalpy responses will bound any other combination of pre-ejected and post-ejected total peaking factors.

The final conservatism employed is the use of bounding values to characterize the many physics and thermal-hydraulic parameters required as input to the point kinetics analysis of the control rod ejection event. This approach is conservative with respect to performing a multi-dimension analysis because the detail (core exposure, cross-sections, etc.) used for the multi-dimension kinetics analysis yields results that are more representative of the event transient response.

The reference made to the comparisons performed by vendors [References 26 and 27] is provided to demonstrate that similar conclusions are obtained for similar applications of reduced-dimension analysis relevant to a multi-dimension analysis. The reference to analyses performed by other utilities [Reference 28] is provided to again demonstrate the reduction of event consequences resulting from the application of a more detailed multi-dimension analysis.

20. Question

How will it be insured that the Reference 26 RCS overpressurization analysis for the rod ejection transient remains valid for future CPSES-1 cycle reloads?

Answer

The intent of the generic analysis presented in Reference 26 was to perform a control rod ejection analysis that would result in a pressure transient that would bound any anticipated control rod ejection event. As such, the generic analysis utilized a very conservative set of assumptions to perform the system overpressurization analysis. Among the more conservative assumptions used in the generic analysis was the selection of an ejected control rod worth that is more than four times the ejected control rod worth assumed in the TU Electric analysis. Be TU Electric will not design and license reload core designs that could produce an event of such magnitude, the overpressurization analysis is not included as part of the control rod ejection analytical methodology.

21. Question

How is the highest worth rod determined for the rod ejection transient?

Answer

The highest worth control rod for the control rod ejection event analysis is determined using a two-dimensional nodal calculation to estimate the ejected control rod worth. If the results of these calculations are inconclusive, i.e., two or more unique control rods have a similar ejected rod worth, a three-dimensional nodal calculation is performed to determine which ejected control rod results in the greatest ejected control rod worth.

22. Question

How do the Section-5.4 CPSES-1 Cycle 1 calculations of the control rod drop event compare to the W predictions?

Answer

As described in Section 5.4 of RXE-91-002, the typical system response to a control rod drop event, as predicted using the TU Electric analytical methodology, is presented in Figures 5.4-3 through 5.4-10. Additionally, Figures 5.4-11 and 5.4-12 depict the comparison between $F_{\Delta H,LIM}$ and $F_{\Delta H,ST}$ for postulated control rod drop scenarios initiated from BOC and EOC conditions, respectively, for CPSES-1 Cycle 1. For each event scenario, $F_{\Delta H,ST}$ is less than $F_{\Delta H,LIM}$, thus satisfying the DNBR acceptance criterion for the event. The system performance of the major RCS parameters, e.g., core power, core fluid temperature, and RCS pressure are analogous to those of the analysis presented in the CPSES-1 FSAR [Reference 11]. The CPSES-1 FSAR does not provide any figures to illustrate that the DNBR acceptance criterion is met for each control rod drop scenario, i.e., figures analogous to RXE-91-002 Figures 5.4-11 and 5.4-12. Instead, the CPSES-1 FSAR states that "In all cases, the minimum DNBR remains above the limit value." As a result, a direct and meaningful comparison of the Westinghouse predicted response to the response predicted using the TU Electric analytical methodology is not practical.

23. Question

Discuss the variation in prompt neutron lifetime of Table 5.5-1. What values were used in the calculations?

Answer

The prompt neutron lifetime for the control rod ejection event analyses presented in Table 5.5-1 varies from 17.5 to 29.0 microseconds. This range is expected to bound all future cycle designs for CPSES. The values presented in Table 5.5-1 represent the value of the prompt neutron lifetime used in the specified analysis.

24. Question

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

Answer

The response to this question will be provided in a separate transmittal.

25. Question

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?

Answer

The response to this question will be provided in a separate transmittal.

26. Question

In the control rod ejection analysis, how is the fission gas release determined from the fraction of fuel melt?

Answer

The fission gas release from the melted fuel is used to derive a portion of the source term needed to determine the offsite radiological dose. The specifications in Appendix B of Regulatory Guide 1.77 [Reference 14], with the limitations specified in the CPSES-1 FSAR, are used to determine the extent and type of fission gas release from the melted fuel. For conservatism, any fuel region that attains or exceeds the fuel melting temperature is assumed to be fully melted. Additional conservatism is included in the offsite radiological dose calculation by assuming that 50% of the iodine contained in the melted fuel is available for release from the plant secondary.

27. Question

In previous analyses, the middle-of-cycle statepoint has been found to be limiting in the evaluation of the control rod drop event. How will it be insured that a middle-of-cycle statepoint is not limiting for future CPSES-1 cycle reloads?

Answer

For cycle exposures at which core physics parameters are not explicitly generated, the core physics parameters are estimated by assuming a linear variation with cycle exposure between the explicitly defined conditions. If a cycle exposure other than those explicitly defined is determined to be more limiting, explicit calculations of the core physics parameters are performed for that core exposure. The core physics parameters at this additional exposure and the values from the original exposure are then used to estimate the core physics parameters at the remaining cycle exposures by again assuming a linear variation with respect to cycle exposure. The process is repeated, as necessary, to confirm that the event-specific acceptance criteria are met for all cycle exposures.

28. Question

How will conservative values of the delayed neutron fraction and prompt neutron lifetime be determined for each of the transients?

Answer

The delayed neutron fraction and prompt neutron lifetime used for the system thermal-hydraulic analysis of the reactivity anomaly events are set in accordance with the results of various sensitivity studies. These sensitivity studies are used to establish the conservative direction, i.e., minimum or maximum parameter values, for each of the specific event analyses.