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SUBJECT: Forwards rev to 940314 response to NRC 931025 RAI re Topical Rept WPPSS-FTS-131, "Applications Topical Rept for BWR Design & Analysis."

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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Docket No. 50-397

August 12, 1994
G02-94-192

U.S. Nuclear Regulatory Commission
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Gentlemen:

Subject: **NUCLEAR PLANT NO. 2, OPERATING LICENSE NPF-21 REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING TOPICAL REPORT WPPSS-FTS-131, "APPLICATIONS TOPICAL REPORT FOR BWR DESIGN AND ANALYSIS" (TAC NOS. M77048 AND M81723)**

References:

1) Letter, dated October 25, 1993, James Clifford (NRC) to J.V. Parrish (Supply System), "BWR Transient Analysis Model (TAC NO. M77048 and M81723)"

2) Letter G02-94-060, dated March 14, 1994, J.V. Parrish to NRC, "Nuclear Plant No. 2, Operating License NPF-21, Response to Request for Additional Information Regarding Topical Report WPPSS-FTS-131, 'Applications Topical Report for BWR Design and Analysis' (TAC Nos. M77048 and M81723)"

Reference 1 requested the Supply System to provide additional information concerning the subject Applications Topical Report. The Supply System response to the request was transmitted to the NRC on March 14, 1994 (Reference 2).

This letter provides a revision to the responses in Reference 2. This revision is made as a result of a telephone conference held between the NRC and its contractor (H. Komoriya of ITS) and the Supply System on June 17, 1994. In addition to the changes discussed, other minor corrections were made to the responses to Question 8. Changes are indicated by revision bars in the left margin.

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If you have any questions, please call me or P.R. Bemis at (509) 377-4027.

Sincerely,

J.V. Parrish (M/D 1023)
Assistant Managing Director for Operations

JVP/slc

Attachment: Revised Response to RAI on WPPSS-FTS-131

cc: LJ Callan - NRC RIV
KE Perkins, Jr. - NRC RIV, Walnut Creek Field Office
NS Reynolds - Winston & Strawn
JW Clifford - NRC
DL Williams - BPA/399
NRC Sr. Resident Inspector - 927N

9408180119

REVISED RESPONSE TO
REQUEST FOR ADDITIONAL INFORMATION
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
TO REVISION 1 OF TOPICAL REPORT WPPSS-FTS-131

Question 1

Generic use of VIPRE-01 for BWR applications was approved with restrictions. As was done for PWR applications of VIPRE, the SER requires each user to submit to the NRC for review and approve a report documenting and qualifying its use with full detailed description. Therefore, the Supply System must submit a separate report (or detailed write-up) documenting its intended use of the code and model qualifications to meet the requirements of the VIPRE SER. Use of CHF correlations other than GEXL must be separately submitted for review and approval (which was done and will be under review) describing use for WNP-2 analysis. Demonstration of their applicability for analysis of mixed cores must also be addressed.

Response

Supply System intends to use VIPRE-01 to evaluate the MCPR for the WNP-2 core at steady-state and transient conditions. Specifically, the following three transients will be analyzed for every reload core unless the specific event can be shown to be non-limiting:

a. Recirculation Flow Controller Failure (Flow Increase Transients from Low Power and Low Flow Conditions)

b. Load Rejection without Bypass (LRNB)

c. Feedwater Controller Failure (FWCF)

Other transients to which VIPRE-01 might be applied are:

a. Turbine Trip without Bypass (normally bounded by LRNB)

b. Other typical non-limiting transients requiring an evaluation of MCPR except for the transients evaluated using steady-state reactor physics methods as discussed in the response to Question 25.

VIPRE-01 MODEL OPTIONS FOR WNP-2

VIPRE-01 contains many user options to select flow correlations, heat transfer correlations, CPR correlations and solution methods. In the Supply System VIPRE-01 Models, they are set as:

1. Flow Correlation

Subcooled Void Correlation: Drift Flux Model[1] (DRFT)
Bulk Void/Quality Correlation: Drift Flux Model[1] (DRFT)
Two-phase friction multiplier correlation: Columbia/EPRI Correlation (EPRI)
Hot Wall Friction Correction: No Correction (NONE)

Note:
1. All empirical coefficients used in the Drift Flux Model are specified by the default or recommended values provided in the VIPRE-01 User's Manual.

2. Heat Transfer Correlation

Single-phase forced convection correlation: EPRI/Dittus-Boelter correlation (EPRI)
Subcooled nucleate boiling correlation: Thom Correlation (THSP)
Saturated nucleate boiling correlation: Thom Correlation (THSP)
Critical Heat Flux Correlation to define the peak of the boiling curve: EPRI-1

Correlation (EPRI)

Transition boiling correlation: Condie-Bengtson Correlation (COND)

Film boiling correlation: Groenveld 5.7 Correlation (G5.7)

3. Correlation for CPR Analysis

ANFB Critical Power Correlation. This is a user-coded correlation, implemented in VIPRE-01 by the Supply System. The ANFB Critical Power Correlation was submitted by SPC to NRC for review and was approved for CPR calculations (Reference 1).

4. Solution Method

RECIRC option

VIPRE-01 APPLICATION FOR WNP-2

All the above selections are based on code developer's verification results (Reference 2) and our own experience gained in performing BWR verification analyses (Reference 3). For licensing calculations, these selections are fixed and procedures are in place to assure that they are not changed without adequate evaluation.

The Supply System VIPRE-01 models (nodalization, correlations, solution method) and hot channel methodology are qualified through benchmark analyses and Supply System to SPC comparisons. Since our models are intended for CPR calculations, the focus of qualification is on how well VIPRE-01 predicts CPR under various conditions. The qualification effort consisted of three stages:

1. Benchmark against steady-state CHF tests: A total of 96 cases were analyzed. VIPRE-01 calculated Critical Powers are essentially identical to SPC's results (See Table B.4-1, Appendix B of the Application Topical Report Revision 1).

2. Benchmark against transient CHF tests: A total of 14 cases were analyzed. The predicted Critical Powers are in good agreement with measured data (See Table B.4-2, Appendix B of the Application Topical Report Revision 1).

3. Comparisons to SPC reload analysis results: Using the Cycle 8 core as an example, three transients were analyzed with VIPRE-01 and compared to SPC's results. The comparisons are discussed below:

   a. Recirculation Flow Controller Failure (See Section 5.3.8 of the Application Topical Report Revision 1)

      It was conservatively assumed that the event was quasi-steady, consistent with SPC's approach. Steady-state calculations were performed with the full core five channels VIPRE-01 model (See Figure B-2, Appendix B of the Application Topical Report). The reduced flow CPR operating limits calculated with VIPRE-01 are in good agreement with SPC's results using the XCOBRA code (Table 5.3.8-2 of the Application Topical Report). Therefore, VIPRE-01 results will provide the same adequate protection for the fuel from flow increase transients as SPC analysis.

   b. Load Rejection without Bypass and Feedwater Controller Failure

      The change in Critical Power Ratio (ΔCPR) for Load Rejection without Bypass and Feedwater Controller Failure events were calculated according to the Hot Channel Methodology described in Appendix B of the Application Topical Report. The Supply System uses RETRAN-02 to perform system simulation providing forcing functions (boundary conditions) for VIPRE-01 whereas SPC uses COTRANSA2 to perform system simulation providing forcing functions for XCOBRA-T (SPC's Hot Channel analysis code). The differences in the forcing

functions inputed to VIPRE-01 and XCOBRA-T make meaningful comparisons to SPC's hot channel calculation difficult. To eliminate the input differences between VIPRE-01 and XCOBRA-T, we requested and obtained the forcing functions (Inlet enthalpy, core inlet and exit pressure, core power, and axial power shape as functions of time) from SPC. Using identical forcing functions, ΔCPRs calculated with VIPRE-01 for SPC 8x8 and 9x9 fuel bundles are compared to SPC's XCOBRA-T results.

ΔCPR CALCULATIONS WITH SPC FORCING FUNCTIONS

Event	VIPRE-01 8x8	VIPRE-01 9x9	XCOBRA-T 8x8	XCOBRA-T 9x9
LRNB(104%P/106%F,EOC8) RPT Operable, TSSS	0.22	0.23	0.21	0.22
FWCF(104%P/106%F,EOC8) RPT Operable, TSSS	0.16	0.17	0.16	0.16

As can be seen from the above table, VIPRE-01 and XCOBRA-T yielded essentially identical results, with VIPRE-01 being slightly more conservative than XCOBRA-T.

For the most limiting transient, Load Rejection without Bypass, additional comparisons (axial void fraction at 0.0 s, 2.5 s and surface heat flux at 0.0 s, 1.2 s, 2.5 s) are shown in Figures 1-1 through 1-5. It is noted that the time of 1.2 seconds used in Figure 1-2 is when the critical heat flux ratio reaches its minimum. Even though the Drift Flux Model in VIPRE-01 predicted a higher void fraction (Figures 1-4 and 1-5), the combination of flow and heat transfer correlations resulted in a predicted surface heat flux that is in good agreement with that predicted by XCOBRA-T (Figures 1-1 through 1-3). Accurate prediction

of heat flux is important in the prediction of CPR.

The VIPRE-01 SER requires the qualification of the Drift Flux Model. Because of limited available transient void fraction data, the Supply System elected to qualify the Drift Flux Model in an integral approach, i.e., applying VIPRE-01 Models with the Drift Flux Model in combination with the flow and heat transfer correlations specified above for MCPR evaluation. MCPR evaluation is the intended use of VIPRE-01 by the Supply System. The conservative predictions in ΔCPR in comparison to SPC's results and the good agreements of benchmark analyses with measured data justifies the use of the Drift Flux Model, integrated with the rest of correlations and options in WNP-2 VIPRE Models, for performing reload analyses.

Based on the steady state and transient benchmarks presented here and in the Applications Topical Report Revision 1, it is concluded that VIPRE-01 predictions of BWR fuel bundle CPR are acceptable for BWR licensing analyses.

The ANFB Critical Power Correlation is applicable for analysis of a mixed core of SPC fuels. The correlation was developed as a generic tool for evaluating critical power and assessing thermal margin for all SPC BWR fuel designs (Reference 1). Differences between specific fuel designs are accounted for by a lattice position dependent function termed F-effective. The correlation uses assembly averaged values of coolant flow, enthalpy, and pressure, including F-effective, to predict the bundle average critical heat flux. The corresponding assembly critical power is determined iteratively and equals the assembly power when the predicted ANFB critical heat flux and the bundle average heat flux are equal. The hydraulic and thermal conditions over which the ANFB corrrelation is applicable bounds the range of parameters over which the plant transients are performed (Reference 1). The mixed core will remain within the ANFB correlation range of applicability. Therefore, the ANFB correlation can be applied to any SPC fuel design bundle, regardless of the mixed core environment.

Use of CHF correlations other than ANFB, due to a change of fuel vendor or fuel design, will be verified by using the same benchmark process as those presented in Appendix B of the Applications Topical Report, Revision 1.

REFERENCES

1. ANFB Critical Power Correlation, ANF-1125(P)(A), ANF-1125 Supplement 1(P)(A) and ANF-1125 Supplement 2(P)(A), April 4, 1990.

2. J. M. Cuta, et at., VIPRE-01, A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCMA, Volume 4, April, 1987.

3. Letter to USNRC from Y.Y. Yung (VMG), Notification of Release and Request for NRC Review of VIPRE-01 MOD-02, February 28, 1990.

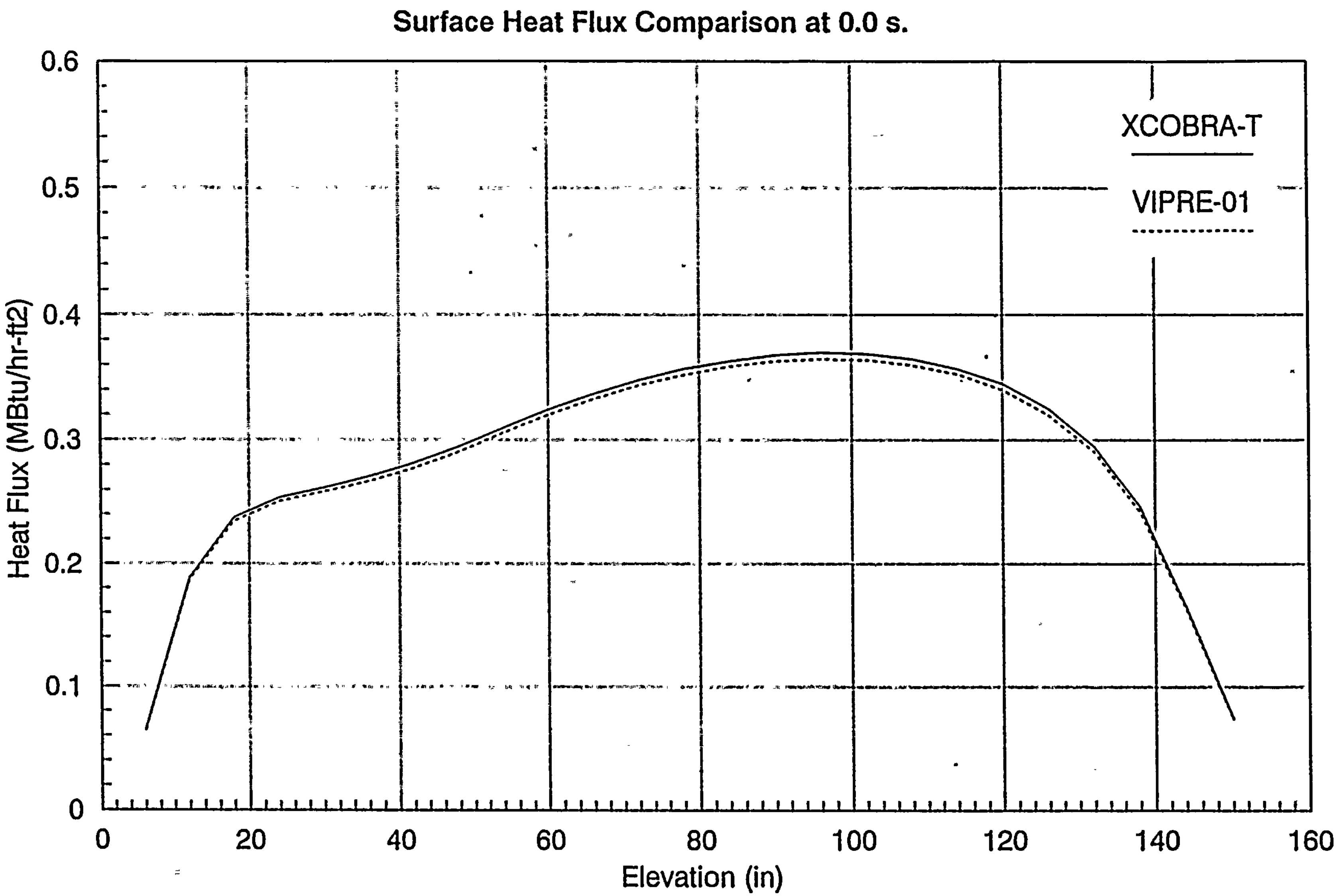

Figure 1-1 LRNB Results, Using SPC Forcing Functions

- 6 -

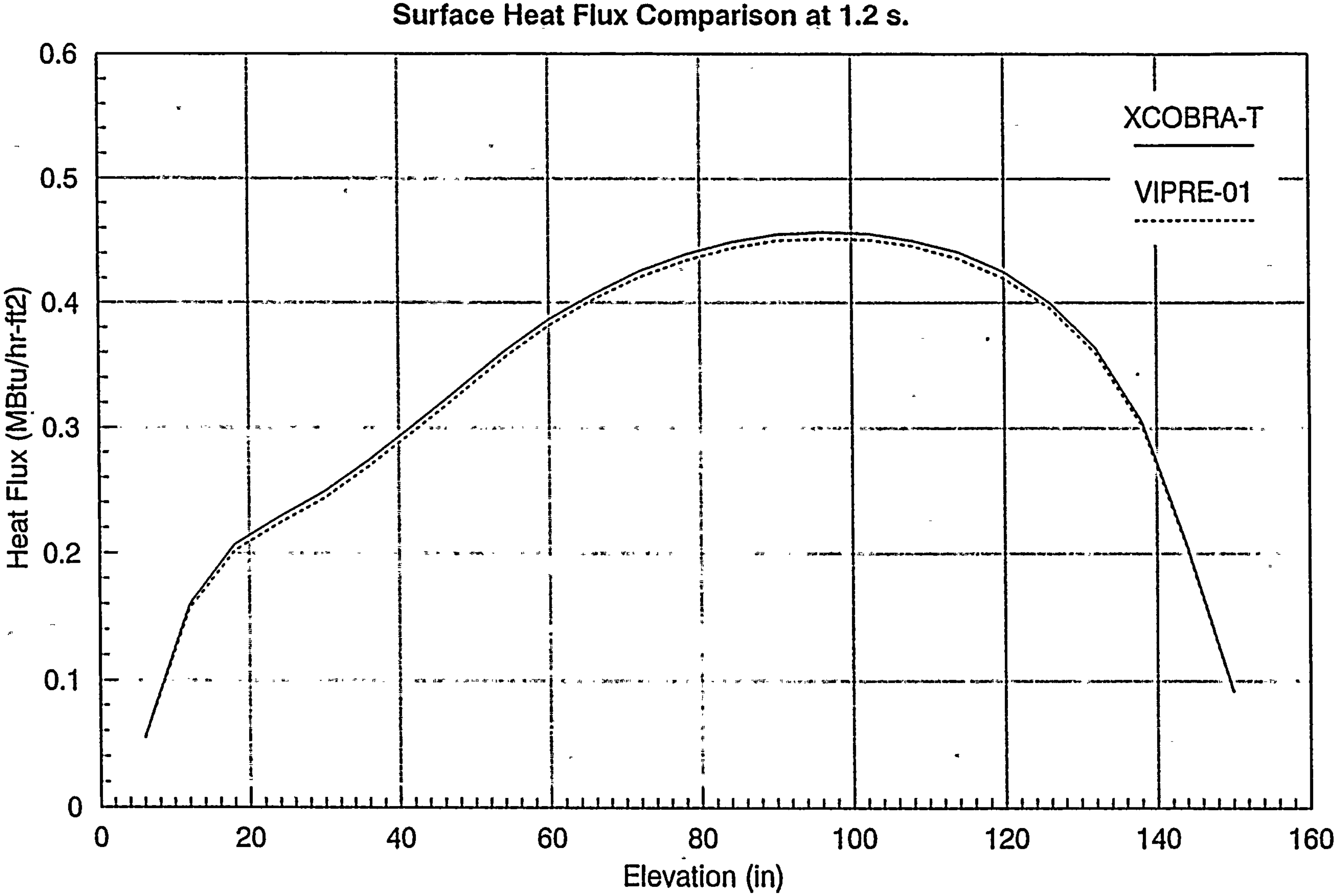

Figure 1-2 LRNB Results, Using SPC Forcing Functions

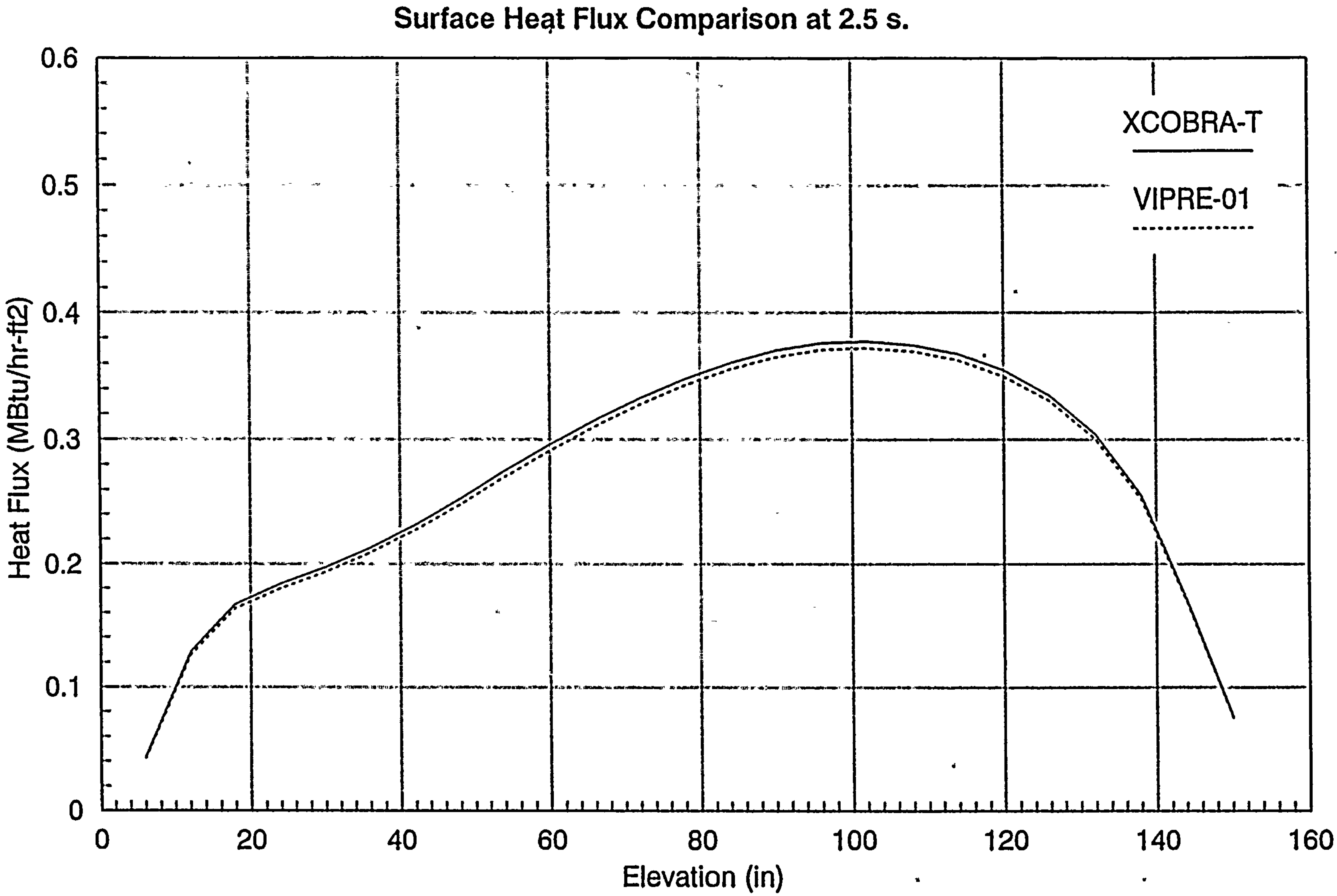

Figure 1-3 LRNB Results, Using SPC Forcing Functions

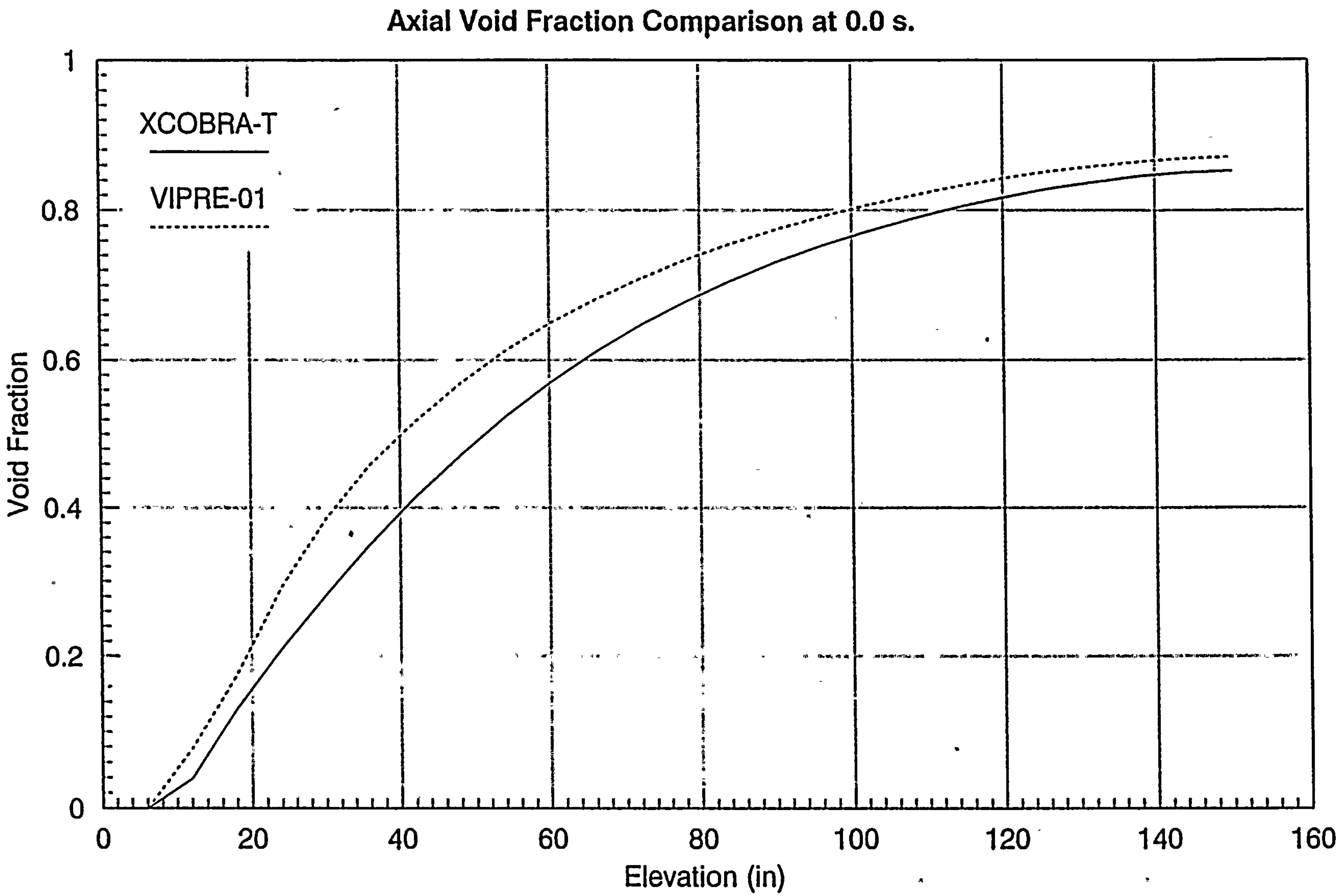

Figure 1-4 LRNB Results, Using SPC Forcing Functions

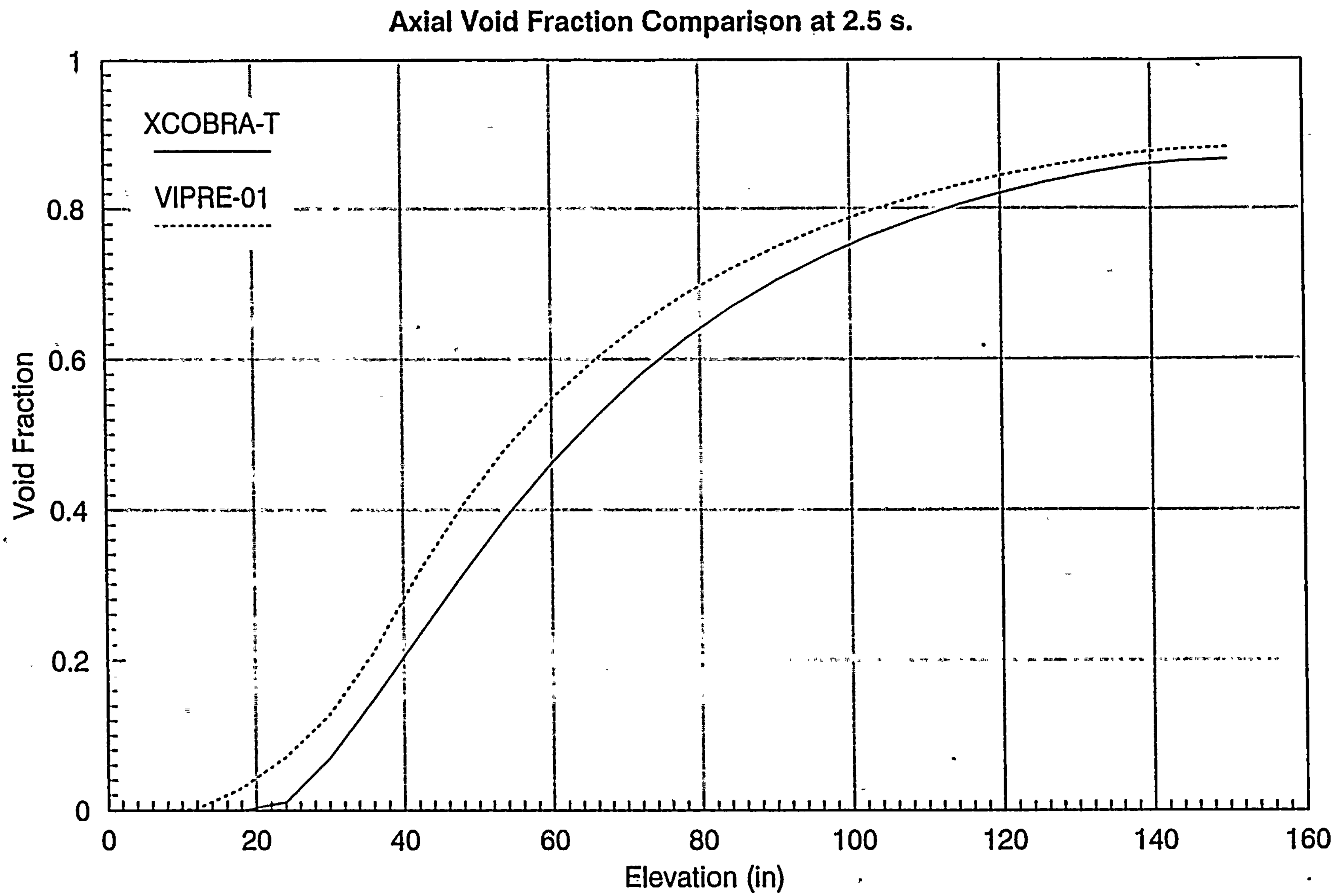

Figure 1-5 LRNB Results, Using SPC Forcing Functions

Question 2

Besides VIPRE and RETRAN, WPPSS uses EPRI developed linked physics codes (CASMO-2E, NORGE-B, SIMULATE-E, SIMTRAN and STRODE) and a set of special purpose codes (ESCORE, FICE, RODDK, TLIM, CALTIP, RBLOCK, and STARS). Since these codes are being used in licensing applications, please identify which of these codes have been reviewed and approved by the NRC and describe ANY local adaptations/modifications (regardless of their impact) implemented in the released versions of these codes. Provide detailed discussion of the impact of changes on the analytical prediction.

Response

The main computer programs used in the Supply System methodology have been described and qualified in Licensing Topical Reports issued by the Supply System and approved by the NRC. The Supply System implementations of CASMO-2E, NORGE-B, SIMULATE-E, and CALTIP are described in the steady state physics topical report (Reference 1), which has been reviewed and approved (Reference 2). The Supply System implementations of RETRAN-02 and SIMTRAN-E are described in the transient modeling topical report (Reference 3), which has been submitted and reviewed (Reference 4). The Supply System implementation of VIPRE-01 is described in the Applications Topical Report Revision 1. ESCORE has been approved by the NRC (Reference 5). RBLOCK (Reference 6) has been reviewed and approved on the Vermont Yankee docket. Information regarding the STARS computer program is contained in the response to Question 3.

All computer programs used in safety analysis applications are subject to the Supply System's Software Quality Assurance program, which is administered as part of the Supply System's approved Quality Assurance program.

All of the externally developed computer programs used in the Supply System methodology were adapted to the Supply System software quality assurance program. Specific modifications

required for compliance include assignment of revision identifiers, access of application-specific traceability routines, and generation of a title page identifying files accessed during the run. Internally developed programs also include these features.

All of the externally developed computer programs in the Supply System methodology were converted for execution on a network of Unix-based workstations (except as discussed in response to Question 3); this activity required changes in file manipulation routines and non-Fortran subprograms for consistent interface with the operating system. Validation of the external programs on the workstation network included confirmation that the revised coding provided results which were consistent with the external distribution coding. In addition to the Unix conversion, the RBLOCK program provided by Yankee Atomic also required a change in input algorithm to allow it to process data from the general data interface file used by TLIM.

The Studsvik of America developed CASMO-2E program was updated under the Supply System Quality Assurance program to include additional geometries for advanced fuel assembly designs. This work was performed under contract by Studsvik of America and is consistent with geometries used by Studsvik in CASMO-3. As mentioned above, this modification was verified and validated under the Supply System Quality Assurance program. The geometry changes in CASMO-2E have been evaluated for the Siemens Power Corporation 9x9-9X Fuel Design in the Core Follow for Cycles 7, 8 and 9 and for the analyses performed for Cycle 8 in the Applications Topical, WPPSS-FTS-131, Rev. 1. The results of these analyses indicate that the WNP-2 cores with 9x9-9X design give consistent uncertainties in nodal power distribution and k-effective when compared to past results where only 8x8 fuel designs were present.

The EPRI-developed RODDK program is not used to perform safety-related calculations; rather, it provides an estimate of control rod worths thereby reducing the scope of the maximum rod worth evaluation required for a core configuration. The safety-related parameters associated with the determination of maximum control rod worth are calculated with SIMULATE-E.

Besides the Unix adaptation of SIMTRAN-E, the calculational process was revised from the

EPRI released version. In validation of the coding, the Supply System discovered a potential improvement in the convergence of buckling factors when adjusting the SIMTRAN-E power distribution to match the SIMULATE-E restart file values. The released algorithm interrupts the iteration whenever a preset limit value is reached, occasionally allowing poor initial value estimates to prevent convergence on a mechanistically correct value within the preset limits. The revised coding allows the iteration to continue outside the limits but restricts the end value so that it falls within the limits. This coding change results in convergence on the SIMULATE-E power distribution for many cases that previously did not converge.

FICE, TLIM, CALTIP, and STRODE were developed internally at the Supply System. A further discription of these codes can be found in the Applications Topical. FICE reads CASMO-2E punch output files, sorts the lattice physics data by nodal state, and calculates rod-centered local peaking factors for the ANF-B critical power correlation. TLIM reads SIMULATE-E restart files and calculates thermal limits based on core state parameters. CALTIP calculates incore instrument responses. STRODE corrects SIMTRAN-E output for thermodynamic differences between SIMULATE-E and RETRAN-02 and adjusts delayed neutron fractions.

REFERENCES

1. B.M. Moore, A.G. Gibbs, J.D. Imel, J.D. Teachman, D.H. Thomsen, and W.C. Wolkenhauer, Qualification of Core Physics Methods for BWR Design and Analysis, WPPSS-FTS-127(-A), Washington Public Power Supply System, March 1990.

2. Letter, James Clifford (NRC) to G.C. Sorensen (Supply System), Evaluation of Topical Report WPPSS-FTS-127 'Qualification of Core Physics for BWR Design and Analysis' (TAC No. M76783), October 23, 1992.

3. Y.Y. Yung, S.H. Bian, D.E. Bush, and B.M. Moore, BWR Transient Analysis Model, WPPSS-FTS-129(-A), Revision 1, Washington Public Power Supply System, September

1990.

4. Letter, James Clifford (NRC) to J.V. Parrish (Supply System), BWR Transient Analysis Model (TAC Nos. M77048 and M81723), October 25, 1993.

5. I.B. Fiero, et al., ESCORE - the EPRI Steady-State Core Reload Evaluation Code: General Description, EPRI NP-5100-L-A, April 1991.

6. J. Pappas, RBLOCK: A Rod Block Monitor Simulator, User's Manual,"YAEC-1509P, Revision 1, Yankee Atomic Electric Company, 1987.

Question 3

Provide a thorough description of the STARS computer code used in connection with the Statistical Core Uncertainty (SCU) methodology.

Response

A description of the STARS computer code (Reference 1) is presented in five sections: (i) STARS Overview; (ii) STARS Theory; (iii) Supply System Usage; (iv) STARS Verification and Validation; and (v) Other Plant Applications.

STARS OVERVIEW

STARS (Statistical Transient Analysis by Response Surface) is a PC-DOS computer code designed to apply the EPRI statistical combination of uncertainties (SCU) methodology to a variety of plant performance and safety analyses (Reference 2). Since it is highly unlikely that all of the event analysis inputs would be simultaneously at their most adverse or design limit values, it is logical to treat the most sensitive parameter(s) in a statistical manner. The SCU methodology provides a mathematically rigorous and computationally efficient way of reducing the sources of unnecessary conservatism in plant analyses.

In STARS, the analysis figure of merit (e.g., RCPR) is treated as a quantity whose statistical properties can be estimated from a combined response surface/Monte Carlo analysis of the results of a systems simulation code or code package, such as RETRAN-02/VIPRE-01. The employment of a response surface greatly reduces the number of simulation code cases required to obtain meaningful statistical results; direct stochastic simulation is often prohibitively expensive. Each response surface is a polynomial approximation to the systems code, for a specific event and a particular analysis result, which is valid over the operating range. STARS can be used to develop second-order response surfaces (with all cross-terms) for up to nine input parameters; it can also be used to establish selected cubic and quartic dependencies. Given an

explicit probability distribution for each of the uncertain input parameters, a Monte Carlo simulation can be employed to obtain the combined effect of the uncertainties in the statistically treated input parameters. By treating only the most sensitive parameters of an analysis statistically, it becomes economically feasible to use STARS to develop an overall probability distribution for an analysis result of a specific event.

STARS was originally created to assist its users in applying the SCU methodology developed in the EPRI LWR setpoint analysis guidelines project (Reference 3). STARS was developed by S. Levy Incorporated under funding from EPRI and the NRC. The computer code has been thoroughly tested and placed under the S. Levy Incorporated Quality Assurance Program. The Supply System is applying the code without any local adaptions or modifications and has placed maintenance of the code under QA procedures for externally developed codes as described in the response to Question 2.

STARS THEORY

The statistical simulation of an event often requires that a large number of cases be run to obtain the desired accuracy. In principle, a probability distribution function (PDF) for the object of the analysis could be constructed by randomly sampling the PDFs for each of these sensitive input parameters and then executing the event analysis code(s) for each set of samples to obtain a series of event analysis results. However, to run these cases using only a systems simulation code (i.e., direct stochastic modeling) could require the expenditure of an unacceptable amount of computational resources.

Response Surface Methodology

Response surface methods provide an economical way of reducing the number of system and core thermal-hydraulic code cases required to make statistically significant statements about the event analysis result (also called the "response").

The basic idea behind a response surface approach is to run a limited number of cases using the systems simulation code to generate output points for the presumed "exact" relationship, $Y = f(X)$, between the output variable of interest (Y) and a given set of N independent input parameters, $(x_1, x_2, x_3,..., x_N)$. Then, least squares fitting techniques are used to determine an analytical expression, Y_{rs}, that can be used as an approximation to Y (within the limited region of interest and for the particular event being analyzed). This simulation code substitute, or response surface, can then be inexpensively evaluated for input parameter samples defined by any Monte Carlo method. Because of the reduced computational cost per sample, the sampling scheme efficiency is relatively unimportant.

The basic input to a response surface calculation is a set of results obtained from the runs of a simulation code (e.g., VIPRE-01). Each case includes a particular combination of input parameter values and the corresponding output variable value predicted by that run. For now, it is assumed that the simulation code computes the analysis result with negligible error. The specific cases to be run will depend on the experimental design of the response surface.

In principle, a least–squares approach will generate a set of coefficients defining the response surface so long as the number of cases, NR, is at least as large as the number of free (or unknown) coefficients, K, in the response surface polynomial. However, the cases must be chosen carefully so that all of the proposed dependencies in the fit are accurately determined. The choice of the particular cases to be run is often referred to in the statistical literature as the experimental design. Furthermore, it is preferable for the fit to be performed with additional simulation code cases to facilitate an error analysis and reduce the fitting error.

Experiments to support multiple linear regression analysis typically start from a factorial design. The number of "levels" refers to the number of values that each independent parameter may assume. A full–factorial design simply includes all possible levels of each free parameter ("factor") in all combinations with all other parameters. Such a design for M levels requires M^N cases. Various "fractional" factorial designs can also be defined to include only part of this large case list, while preserving the important dependencies in the regression equation. Thus,

the number of cases that must be run using the systems analysis code depends on the experimental design.

The upper and lower values of the free parameter are typically selected at plus or minus two standard deviations (σ) from the nominal value (μ). A two-level design can provide no information about quadratic terms, but it does allow evaluation of cross-terms between parameters. A two-level design is often called a first-order design because only linear and bilinear (interaction) terms can be represented in the response surface. To allow for the treatment of quadratic terms, a full-factorial design would require three levels and thus 3^N cases. Although the case list is manageable for three parameters, it becomes very large for situations involving more parameters. For four, five or six parameters, a total of 81, 243 or 729 cases would be required.

A more economical experimental approach to determining a second-order fit can be achieved by using an Orthogonal Central Composite Design (OCCD) described in the text by Myers (Reference 4). Central composite designs are two-level, full-factorial designs (ordinarily suitable only for linear and bilinear relationships) that are augmented by additional cases required for estimating quadratic coefficients. The two-level design of 2^N cases is augmented by a central point having nominal parameter values and by 2N axial points, for which only one parameter at a time is varied. Thus, the number of cases required for an OCCD is only $2^N + 2N + 1$. For three parameters, this reduces the number of analysis code cases required from 27 to only 15. For 4, 5 or 6 parameters, the number of cases required would drop dramatically from 81, 243 or 729 to the more manageable 25, 43 or 77. Thus, an OCCD approach offers a sufficiently accurate representation of the response surface with only a limited expenditure of computational resources.

The 2N axial points for each parameter, $\mu \pm 2A\sigma$, will supply values more extreme than $\mu \pm 2\sigma$ for $N > 2$; this ensures that the response surface will be used for interpolation of parameter samples, rather than for extrapolation. The parameter 2A is defined as:

$$2A = \left[\sqrt{2^{N+2} \left(2^N + 2N + 1 \right)} - 2^{N+1} \right]^{1/2}$$

Note that, for two parameters, the OCCD design reduces to simply a full-factorial design (i.e., $NR = 2^N + 2N + 1 = 3^N = 9$ and $2A\sigma = 2\sigma$ for $N = 2$). For $N = 1$, the $\mu \pm 2A\sigma$ values supply additional data points used to reduce the fitting error.

The STARS fit may be developed in terms of normalized (non-dimensionalized) independent variables:

$$\eta_i = (x_i - \alpha_i)/\beta_i$$

where α_i and β_i are user-input constants. For the i^{th} parameter, α_i should be equated to the nominal value of that parameter, so that $\eta_i = 0$ for the nominal case. The parameter β_i should correspond to the expected range of the random input variable. For example, β_i can be approximated by $2\sigma_i$, where σ_i is the standard deviation of the i^{th} input parameter. Note that the original variables are preserved if $\alpha_i = 0$ and $\beta_i = 1$ for each parameter i.

A second-order response surface may be defined in terms of the normalized independent variables, η_i, by determining the free coefficients B_0, B_i, B_{ii}, and B_{ij} such that the output can be predicted by the form:

$$Y_{rs} = B_0 + \sum_{i=1}^{N} B_i \eta_i + \sum_{i=1}^{N} B_{ii} \left(\eta_i^2 - C \right) + \sum_{i=1}^{N} \sum_{j=1,\, j<i}^{N} B_{ij} \eta_i \eta_j$$

where C is a known normalization coefficient used to provide orthogonality in the basis polynomials to ensure that the estimates for the regression coefficients are not correlated. For $N \geq 2$, C is defined as:

$$C = [\, 2^N/(2^N + 2N + 1)]^{1/2}$$

Uncorrelated regression coefficients can provide a means of ranking sensitivity effects. By definition, the coefficient C serves no purpose for those response surfaces having a single independent parameter. The B_{ii} C term essentially gets lumped into the B_0 constant term.

The coefficients of the response surface model may be estimated using the methods of linear regression. In particular, the coefficients are chosen in the traditional least squares sense to minimize the standard error (SE) of the fit:

$$SE = \sqrt{\frac{1}{NR - K - 1} \sum_{j=1}^{NR} \left(Y_{rs,j} - Y_j \right)^2}$$

where K = (N + 1) (N + 2) / 2 is the total number of undetermined coefficients in the fit.

STARS can handle up to 350 data points in evaluating the response surface polynomial fitting coefficients. The least squares evaluation of the free coefficients employs established public domain algorithms (Reference 5) based upon the singular value decomposition (SVD) method. The SVD method uses an extremely robust approach to avoid numerical roundoff problems.

Statistical Analysis of Response Surface

For any set of parameter values (x_1, x_2,..., x_N), the polynomial fitting coefficients define the "response" Y_{rs} (x_1, x_2,..., x_N), i.e., the approximation to the "true" dependent variable Y that would be predicted for that case by the event analysis code(s).

The next step is to use this response surface approximation to perform a statistical analysis of the event analysis result. For a linear response surface with normally distributed input parameters, the output distribution is also normal and can be derived analytically. Many applications, however, require a second-order treatment. In general, the dependent variable

distribution cannot be predicted analytically. The statistical properties of the dependent variable, however, can be obtained by a Monte Carlo analysis of the response surface. Monte Carlo, or stochastic simulation, methods involve the use of repeated computer trials to sample values of the independent parameters of a dependent variable in order to draw inferences about this variable. The statistical characteristics of each input parameter must be explicitly defined.

In STARS, the Monte Carlo analysis is carried out through the evaluation of a number of "histories", i.e., particular combinations of each of the N independent parameters. For each history, a particular value for each input parameter is selected randomly, using that parameter's cumulative probability distribution. In the same manner, STARS also randomly samples the normally distributed fitting error, ϵ, of the response surface. For each history, denoted by index j, the response surface is then evaluated to obtain a particular value Y^j for the dependent variable:

$$Y^j = Y_{rs}\left(X_1^j, X_2^j, \ldots\ldots, X_N^j\right) + \epsilon^j \qquad for \quad j = 1, 2, 3, \ldots. J$$

where the number of histories J will typically be in the range of 1,000 to 100,000. Experience indicates that 100,000 histories are more than sufficient to accurately quantify the probability of the event analysis result. The J samples of the dependent variable Y (e.g., RCPR) are then analyzed to obtain the cumulative distribution function, G(y), for the dependent variable. Formally:

$$G(y) = \hat{P}(Y \leq y) = \frac{1}{J} \sum_{j=1}^{J} U(y - Y^j)$$

where U is the unit step function such that $U(z) = 1$ for $z \geq 0$ and $U(z) = 0$ for $z < 0$. Thus, G(y) is simply the fraction of histories for which the dependent variable Y is less than or equal to y.

The Monte Carlo simulation process in STARS employs RAN1, a random number generator developed by Knuth (Reference 5). RAN1 returns a uniform random deviate between zero and

one. It is machine independent and based on three linear congruential generators and a shuffling routine; each generator uses modular arithmetic and recurrence relations to produce statistically random sequences of numbers. A seed is initialized to an arbitrary integer before the first call to RAN1. The same initializing value of the seed will always cause RAN1 to return the same random sequence, regardless of machine; this feature is useful when trying to replicate the results of prior runs.

STARS uses these random numbers (between 0 and 1) as table look-ups in the cumulative distribution function (CDF) that is computed for each independent parameter, based on its input data. The value of the parameter that corresponds to the value of its CDF chosen at random becomes the particular history of the input parameter distribution being sampled; up to 100,000 histories are allowed. The response surface for the output variable is then evaluated using those samples. After all the histories have been gathered, STARS compiles the statistics of the PDF for the output variable.

Accounting for Model Uncertainties

Thus far, it has been assumed that the results of the model (e.g., event analysis code) used to generate a response surface are known with complete certainty. However, in practical applications, this is usually not the case. Model uncertainties exist due to physical model approximations, numerical solution techniques, plant process model simplifications, and the like. Hence, the final task is to analytically combine the response surface distribution with the overall model uncertainty. To determine the model uncertainty, analysis models are run with perturbed inputs and modeling options to quantify the effect of these changes on the event analysis results. The results of the model uncertainty studies should then be statistically combined to characterize the overall model uncertainty.

STARS provides its users with the option of "convoluting" the response surface distribution with the model uncertainty distribution in order to establish an overall probability distribution for the event analysis result. The convolution operation effectively smooths the response surface

distribution consistent with the model uncertainty distribution. The resulting overall probability distribution is used to assess compliance with the event acceptance limit.

For example, given a uniform model uncertainty over a finite range, the effect of convolution would be to smooth the discrete output distribution by taking an average of each of the points within the range and assigning that value to the range midpoint. The same would then be done for each point within the STARS output variable PDF. More typically, the model uncertainty might be a peaked function that asymptotically falls to zero in both directions away from its maximum that occurs at zero. An example might be a normal distribution with a zero mean and some standard deviation.

In either case, the effect of convolution would be to smooth the original STARS output PDF by taking a weighted average of the surrounding points. The convoluted PDF would have a lower peak and larger tails as compared to the original PDF. Furthermore, the outlying values in the convoluted PDF would have higher probabilities than the corresponding values in the original PDF. And, since one or both extremes of an outcome distribution are generally the most important from an event consequences standpoint, convolution results in a more conservative distribution.

Mathematically, convolution is defined in this manner. Given two function g(t) and h(t), we can define the convolution of those functions, denoted by g * h, as:

$$g * h = \int_{-\infty}^{\infty} g(T)\, h(t - T)\, dt$$

where g * h is also a function of t. It can be shown that the Fourier transform of the convolution of two functions is simply the product of their individual Fourier transforms.

In STARS, a Fast Fourier Transform (FFT) algorithm is used to compute the discrete Fourier transform of the STARS output variable PDF and of the model uncertainty PDF. The two

transforms are multiplied together, component by component. The FFT algorithm is then used to take the inverse discrete Fourier transform of the products. Additional information on convolution and Fourier transforms can be found in Reference 5, from which the STARS convolution routines were adapted.

SUPPLY SYSTEM USAGE

The Supply System analysis employs a second-order response surface with one independent parameter (Reference 6). For the WNP-2 generator load rejection without bypass event, only the control rod scram time was treated as an independent variable and the response surface assumed the simple form:

$$\mathrm{RCPR} = B_0 + B_1 T + B_{11} T^2$$

As shown in Tables A–4 and A–5 of Reference 6, five different scram times (at times -2σ, -1σ, 0, $+1\sigma$ and $+2\sigma$ relative to the mean scram time) were analyzed for each fuel type in order to construct the response surface for the generator load rejection without bypass event. The RCPR evaluated for each of these cases was input to the STARS code. STARS then calculated the fitting coefficients B_0, B_1, and B_{11}. The response surface provided an excellent fit to the calculated RCPRs as measured by the relatively small RMS fitting errors for the RCPRs of both fuel types.

The Supply System then evaluated each response surface over the course of 100,000 trials. This generator load rejection event simulation yielded a probability distribution for the RCPR that included scram time uncertainty and response surface fitting error, but did not account for any uncertainty in the RETRAN-02/VIPRE-01 model itself.

The final task was to account for the model uncertainty in the analysis codes by employing the STARS convolution option. The Supply System employed a model uncertainty with a normal distribution having zero mean and a standard deviation equal to one-half of the model uncertainty

(in terms of RCPR). From Table A-1 of Reference 6, the 2σ value for the 8x8 fuel overall model uncertainty was 0.0267. Hence, a normal distribution with zero mean and $\sigma = 0.0134$ was input to STARS. Table A-2 indicates that the 9x9-9X fuel had a model uncertainty of 0.0291 and hence a STARS convolution input of $\mu = 0$ and $\sigma = 0.0146$. A convolution of the response surface and model uncertainty probability distributions yielded a distribution for the overall probability of the event analysis result (RCPR). This probability distribution was then used to establish the core operating limit minimum critical power ratio (MCPR).

STARS VERIFICATION AND VALIDATION

STARS performs three major types of calculations: response surface construction; statistical (Monte Carlo) analysis of the response surface; and accounting for model uncertainties. Since STARS uniquely combines several analysis functions, there is no single publicly available data set or code against which to benchmark all of STARS at once. However, the three main functions of STARS have been tested both individually and in various combinations (Reference 1). A brief description of the verification and validation of STARS follows.

Evaluation of the response surface construction was rather straightforward. First, polynomial equations were arbitrarily conceived to represent exact relationships. Then, a spreadsheet program was used to evaluate these equations for sets of randomly selected data. These results were used to build 9 STARS input files. In all test cases, STARS generated curve fits with very small standard (RMS) errors and pointwise residuals. For first and second order polynomials, both with and without cross-terms, STARS generated curve fits whose coefficients were very close to those of the original spreadsheet equations. This conclusion was demonstrated for up to the maximum of nine input parameters allowed by STARS.

Three types of tests have been performed to verify the Monte Carlo analysis process. First, STARS was run with one normally distributed parameter and a set of data points that implied $Y = X_1$. With this input, STARS should produce a probability distribution for the dependent variable that, allowing for sampling and round-off errors, is an exact replica of the PDF for the

single input parameter. STARS evaluated the response surface over 100,000 random trials and yielded dependent variable PDF whose statistics corresponded almost exactly to the statistics of the normal distribution that was used as input.

Next, STARS was employed to simulate the rolling of two ordinary playing dice. Each input parameter was modeled by a CDF that corresponded to the probability of rolling a 1, 2, 3, 4, 5 or 6 by defining an input table containing steps of probability 1/6 in the narrow intervals bounding each of the six possible input values. Four sets of data points were entered that implied $Y = f(X) = X_1 + X_2$, or the sum of two dice. By fitting a simple linear response surface and then evaluating it over the course of 100,000 trials, STARS confirmed the existence of a "lucky seven" and calculated a probability distribution for the sum of the two dice whose statistics corresponded very closely to the exact analytical solution.

Lastly, STARS was used in an attempt to reproduce the response surface/Monte Carlo calculations in a topical report submitted to the NRC for the RCPR of a generator load rejection event in a boiling water reactor (Reference 7). The analysis employed a full-factorial, second-order design for two parameters (steam flow and scram speed), augmented by eight additional data points for all the possible input combinations with $\mu_i \pm \sigma_i$. STARS was able to almost exactly reproduce the results of that submittal; in fact, the STARS response surface for RCPR had a smaller fitting error than did the actual submittal to the NRC. These three tests demonstrated that STARS can sample from a variety of parameter distributions and use those samples to accurately evaluate both linear and non-linear response surfaces.

The convolution routines in STARS were broken out separately and tested with a driver program in order to test simple problems that could be solved analytically. For instance, the convolution of two rectangular (uniform) distributions should result in a trapezoidal distribution. Given the appropriate inputs, the driver program for the convolution routines produced such a distribution, although with the "white noise" that one would expect as a result of performing multiple Fourier transform manipulations. Similarly, the convolution routines were used to successfully shift a probability distribution up and down the axis of its independent parameter, according to the rule

supplied by the model uncertainty PDF. The model uncertainty routines were also able to accurately generate the normal or uniform PDFs that can be requested by the user.

OTHER PLANT APPLICATIONS

The SCU methodology and the STARS code have been applied to address a wide range of setpoint and equipment performance concerns associated with the plant safety or performance analysis. The application of STARS at three different nuclear plants is described below.

OYSTER CREEK (GPU NUCLEAR)

An undesirable overlap had been identified in the Oyster Creek high-pressure setpoints when accounting for measurement uncertainties experienced during plant operation. The SCU process (Reference 3) was employed to provide increased margin for measurement uncertainties. The primary goal of the plant performance requirements was to avoid the opening of a safety valve that leads to the unnecessary discharge of radioactive steam to the primary containment. To satisfy this plant performance criterion, it was necessary to demonstrate that the event analysis consequence did not exceed the limit. Through the use of a mixed SCU method, it was possible to provide a comprehensive justification for the desired technical specification changes to the high-pressure setpoints.

STARS was employed to develop response surfaces to simulate RETRAN-02 peak pressure calculations for the limiting transient events (Reference 8): (i) turbine trip with bypass and (ii) main steam isolation valve closure. Next, STARS was used to perform a Monte Carlo simulation (with 100,000 trials) to obtain the aggregate effect of the uncertainties. Finally, using the MOONS setpoint overlap analysis computer program (Reference 9), the probability distribution computed by STARS for the peak pressure was combined with the distribution for the lift pressure of the first-opening safety valve computed by the PLANETS network analysis computer program (Reference 10) to yield the probability of opening a safety valve and thereby violating the plant performance goal.

THREE MILE ISLAND-1 (GPU NUCLEAR)

To order to modify the TMI-1 flux/flow trip setpoint, it had to be demonstrated that the proposed setpoint met the regulatory and licensing requirements. To satisfy this requirement, the limiting design basis events affected by the flux/flow setpoint change were identified. The controlling safety criterion for the flux/flow setpoint is entirely based on departure from nucleate boiling ratio (DNBR) protection. The design basis events were analyzed using the RETRAN-02 and VIPRE-01 computer codes to calculate the minimum DNBR. To limit the uncertainty analysis to a manageable process, the most sensitive parameters were determined. A mixed SCU process was employed to reduce the VIPRE-01 caseload requirements while obtaining most of the available reduction in conservatism associated with a pure SCU process.

Response surface algorithms were developed to approximate the VIPRE-01 DNBR calculations for the limiting events over the range of uncertainties in the sensitive parameters (Reference 11). An orthogonal central composite design was used. A Monte Carlo simulation was then performed to obtain the aggregate effect of the uncertainties. Finally, the probability of exceeding the DNBR limit was calculated; it showed that the proposed change in flux/flow setpoint satisfies the event acceptance limits for the design-basis events. The STARS code was employed to obtain the response surface fitting coefficients. Next, STARS was used to perform 100,000 Monte Carlo samples of the response surface, based on the probability distributions for the two or three sensitive parameters, depending on the particular event signature. Based on the DNBR probability distribution yielded by STARS, the proposed change in the flux/flow setpoint was shown to satisfy the event acceptance limit for the design basis events.

VERMONT YANKEE (YANKEE ATOMIC)

Yankee Atomic Electric Company (YAEC) anticipated a reduction in the operating margin associated with longer operating cycles at the Vermont Yankee Nuclear Power Station. The analysis methods used to demonstrate the existing operating margin, employed a point kinetics model with conservative inputs. YAEC employed the RETRAN-02 code to analyze limiting

transients and to establish MCPR limits for each Vermont Yankee fuel reload. YAEC staff felt that, by applying the more accurate one-dimensional kinetics model option available in RETRAN-02, enough improvement in the MCPR operating margin could be obtained to offset the anticipated margin reduction.

The calculation, however, had to contain enough conservatism to satisfy licensing requirements when one-dimensional kinetics was used in place of the point kinetics model. The use of the one-dimensional kinetics model resulted in the expected gain in the transient MCPR margin. To demonstrate an acceptable level of conservatism in the determination of the MCPR operating limit, YAEC used the EPRI SCU methodology (Reference 12) to calculate the combined uncertainty due to model and scram-speed uncertainties. To establish the scram-speed uncertainty in the MCPR margin calculations, a response surface was constructed by means of the STARS code. The uncertainty in the underlying RETRAN-02 model was determined by means of sensitivity studies. The resulting MCPR margin was calculated by convoluting the probability distributions for scram-speed and model uncertainty.

YAEC demonstrated that its proposed licensing approach, which is based on RETRAN-02 one-dimensional kinetics, is adequately conservative. This determination was made by comparing the CPR operating limits with the result of the statistical evaluation using STARS. The NRC approved the utility's revised methodology in December 1989.

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Question 4

Please identify the transients for which the statistical uncertainty methodology will be used. In addition, please provide the following: (1) identification of the range of parameters used in the model development (not just for the generator load rejection event); (2) justification that the applicable ranges encompass the conditions expected to be encountered in all relevant transient analyses; and (3) justification of the uncertainty distribution function assumed for each.

Response

The Statistical Combination of Uncertainties (SCU) methodology will be used for the limiting system transient event calculated by RETRAN-02 and VIPRE-01. Potential limiting events are rapid pressurization events which include generator load rejection without bypass, turbine trip without bypass, and feedwater controller failure - maximum demand. For the Cycle 8 analysis shown in the Applications Topical Report Revision 1, the limiting transient is generator load rejection without bypass and this is the event for which SCU methodology was used.

The specific parameters used in the model development and qualification process are identified in WPPSS-FTS-129 Rev. 1 "BWR Transient Model". For comparison to the Peach Bottom Unit 2 turbine trip test data, best estimate model inputs for the specific test conditions were used. In the reload fuel safety analysis process, a combination of best estimate and deterministic (license basis) inputs were used. The license basis inputs are selected to conservatively bound the range of that parameter for the specific event analyzed. These license basis inputs are defined in WPPSS-FTS-129 Rev. 1. The remaining parameters are taken at their best estimate values. The ranges of the best estimate parameters that are subject to potentially significant uncertainties are covered by the statistical overall model uncertainty analysis described in Appendix A of the Applications Topical Report Revision 1. Therefore, in this process, the treatment of uncertainties covers the range of expected conditions associated with the potentially limiting events. The remaining best estimate parameters are considered to be well known and are not subject to significant uncertainties.

The ranges of parameters used in the model development were selected to ensure that they bound the uncertainty ranges expected in the WNP-2 plant, independent of transient types. For the most sensitive parameters, in terms of RCPR, in the model uncertainty determination (i.e., void coefficient and prompt moderator heating, see Tables A-1 and A-2 in the Applications Topical Report, Revision 1), the ranges are deliberately biased on the conservative side. In the case of void coefficient, an EPRI study (Reference 1) for WNP-2 indicated that the void coefficient has an uncertainty range (2-standard deviation) of 6.94%. For the licensing application, a value of 13% is used, leading to a conservative bias of 6.06%. For the case of prompt moderator heating, based on a GE analysis for a BWR/4 plant (which uses similar fuels as in the BWR/5 plants, see Reference 2), a 95% confidence value was determined to be 20%. For conservatism, the Supply System used a value of 25% in our licensing analysis, leading to a conservative bias of 5%. For the rest of the parameters, as discussed in the Applications Topical, the majority of them are based on a generic NRC review of a GE analysis (Reference 3), which used a data base covering various plant types. These data analyses were performed independent of the transient types. Given the fact that they are generic in nature and are reviewed and approved by the NRC, they are adequate for the licensing applications to WNP-2. The remaining parameters in Table A-1 and A-2 in the Topical are based on the studies by TVA and GPU (see discussions in Appendix A of the Topical for identification of these parameters). Even though they are based on plants which are not identical to WNP-2, their application to our analysis is judged to be acceptable based on individual evaluation of the parameters. For instance, in the case of vessel dome and steam line volumes, an uncertainty value of 5% was judged to be adequate because these parameters are well defined parameters and a small value in uncertainty is acceptable. In fact, GE's analysis (Reference 2) allowed no uncertainties for these parameters. For the Dittus-Boelter Correlations and the Hancox-Nicoll correlations used in the heat transfer calculations, even though they were referenced from the GPU submittals as discussed in Appendix A of the Topical, the uncertainties are based on the fundamental heat transfer phenomenon and are applicable to both plants.

The uncertainty ranges which are derived independent of the particular transients as discussed above are used in the determination of the overall model uncertainties through the RETRAN-

VIPRE calculations using the limiting transient, i.e., load rejection without bypass (LRNB). This process is applicable to the other potentially limiting pressurization transients, e.g., Turbine Trip Without Bypass and Feedwater Controller Failure. This is because the LRNB yields the most severe ΔCPRs and the transient phenomena of other potentially limiting transients are essentially identical to LRNB (i.e., rapid pressurization) but not as severe. Therefore, the overall model uncertainties presented in the Applications Topical are applicable to other potentially limiting pressurization transients.

Uncertainty ranges of the parameters used in the SCU analysis cover the effects of the mixed core. This is justified for the following reasons. First of all, the processes of obtaining the "effective" parameters for the mixed core as described in the response to Question 16 do not introduce additional uncertainties. In addition, as discussed above, the core thermal-hydraulic parameters and neutronic parameters are based on the upper bound (i.e., either 95% confidence or 2-standard deviation) values accounting for different fuel types. The uncertainties for the system parameters (i.e., recirculation systems, steam line models, and vessel and loop geometries) are not impacted by the mixed core configuration. Therefore, the model uncertainty and SCU analysis given in the Applications Topical apply to the mixed core configuration in WNP-2.

It is noted that all of the parameters referenced above have been reviewed and approved by the NRC (References 1, 2 and 3).

A normal distribution is assumed for all parameters statistically treated in the Supply System's SCU methodology. This assumption is considered adequate in the licensing model for the following reasons. The parameter used to establish the Response Surface is the control rod scram time. In the statistical analysis to determine the mean and standard deviation, the entire data base of 4858 scram time measurements based on full-core scrams up to the time of the SCU analysis was used. The scram timing is a measurable physical process, and the measurements were taken independently. In addition, the application of the scram timing statistics to the RETRAN model is to represent the 185 control rods by an average single rod in the 1-D core model. The Central

Limit Theorem of statistics indicates that the distribution of the average of N random variables of any distribution is nearly normal for the sampling size as small as N=5 provided the random variable distribution is not extremely skewed (Reference 5). Since the plots of the data indicate that the distributions of measured times for individual rods at four notch positions (particularly at Notch 39, which determines the power turn-around in a transient) are closely represented by a normal distribution and, since the effective sampling size is N=185, which is much larger than 5 as mentioned above, use of normal distribution in the statistical treatment is justified.

For all other parameters used in the model uncertainty analysis, the assumption of normal distribution is sufficient for the following two reasons: First, the model uncertainty as a result of the combined uncertainties from individual parameters is applied to a licensing model which has a large degree of conservatism already established through the use of bounding values for the parameters not treated statistically. Secondly, for the most dominating parameters in the uncertainty analysis (i.e., void coefficient and prompt moderator heating as discussed above), in addition to the assumption of normal distribution, we have applied a conservative bias to these parameters, i.e., a 6.06% bias for the void coefficient and a 5% bias for the prompt moderator heating. Based on these two reasons, the statistical treatment of the uncertainty parameters based on the normal distribution is judged adequate and the overall model is conservative. This approach is consistent with the approved statistical approach used by GE (Reference 2) and TVA (Reference 4).

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Question 5

Provide an NRC approved reference to support the position that no more than four parameters are necessary for development of a response surface.

Response

The number of parameters necessary for development of a response surface is not fixed at four but depends on the number of independent variables in the response surface.

The response surface used by the Supply System for Statistical Combination of Uncertainties (SCU) is of the general form

$$Y_{rs} = B_0 + \sum_{i=1}^{N} B_i \eta_i + \sum_{i=1}^{N} B_{ii} \left(\eta_i^2 - C \right) + \sum_{i=1}^{N} \sum_{j=1, j<i}^{N} B_{ij} \eta_i \eta_j$$

where:

B_o, B_i, B_{ii}, and B_{ij} are response surface fitting coefficients;
η_i are the response surface independent variables;
N is the number of independent variables;
C is a known normalization coefficient that provides orthogonal polynomials;
Y_{rs} is the response surface dependent variable

K, the number of fitting coefficients necessary for development of the response surface, is determined by the number of independent variables N;

$$K=(N+1)(N+2)/2$$

or, in tabular form

N	1	2	3	4	5	6	7
K	3	6	10	15	21	28	36

The response surface fitting coefficients are chosen in the traditional least squares sense to minimize the standard error (SE) of the fit:

$$SE = \sqrt{\frac{1}{NR - K - 1} \sum_{j=1}^{NR} (Y_{rs,j} - Y_j)^2}$$

where:

Y_j = calculated value of the dependent variable

$Y_{rs,j}$ = dependent variable value calculated by the response surface fitting coefficients

NR = number of calculated points used to develop the response surface given by

$$NR = 2^N + 2N + 1$$

This response surface methodology is described in Reference 1 (Same as Reference A6 of the Applications Topical Report Revision 1 except the publishing dates, the earlier date is S. Levy Inc. publishing date, the later date is EPRI's publishing date) and in Reference 2. In theory, any number of independent variables can be included in the development of the response surface. However, the number of cases to be run necessary to achieve an acceptably small fitting error becomes prohibitive for a large number of independent variables. Thus, a practical upper limit for the number of independent variables is in the range of 4 to 6 and is dependent on the benefit that can be achieved at the expense of additional analyses.

For the Supply System application of the Statistical Combination of Uncertainties Methodology to the generator load rejection without bypass event, only the control rod scram time T is used as an independent variable and the dependent variable is RCPR. See the response to Question 26. Therefore, for this application:

$N = 1;\ K = 3$

$RCPR = B_o + B_1T + B_{11}T^2$

NR = 5

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Question 6

For the parameters which are not directly measurable (there are the parameters which are analytically determined, therefore code dependent, such as the nuclear parameters), describe the method by which the "nominal" values are determined and justify the uncertainty ranges considered.

Response

There are basically two types of inputs that are used in the reload fuel safety analysis process: (1) cycle independent parameters and (2) cycle dependent parameters. The cycle independent parameters are those analysis inputs that describe the plant configuration and operating characteristics that do not change from cycle to cycle. The cycle dependent parameters are those parameters that change due to the introduction of the reload fuel assemblies.

For RETRAN-02 inputs, the cycle independent parameters that are included in the WNP-2 model are identified and quantified in WPPSS-FTS-129, Rev. 1, "BWR Transient Analysis Model". These inputs are: (1) the model geometry, including control volumes, junctions and heat conductors, vessel internals, core region, recirculation loops, and steam and feedwater lines; (2) the component models, including jet pumps, recirculation pumps, steam separators, safety/relief valves, and core hydraulics; (3) trip logic for the reactor protection system and for pump and valve actuations; (4) the feedwater and pressure control systems; and (5) the steady state initialization parameters, including the reactor dome pressure, core inlet enthalpy, core flow, recirculation flow, jet pump suction flow, and feedwater and steam flows. The values for the cycle independent parameters used as RETRAN-02 inputs are provided in WPPSS-FTS-129, Rev. 1.

The remainder of the RETRAN-02 inputs are the cycle dependent parameters including the RETRAN-02 kinetics parameters and the core and fuel geometry. RETRAN-02 has the capability of performing analyses using either point or one-dimensional kinetics; however, for

potentially limiting events, one-dimensional kinetics are used in the reload fuel safety analysis process to establish the required core operating limits. The analysis process determining the one-dimensional kinetics parameters for each reload is described in WPPSS-FTS-129, Rev. 1 and in Appendix C of the Applications Topical Report Revision 1. The specific values used in the analysis are dependent on the characteristics of the reload fuel and core design. The treatment of core and fuel geometry for reload cores is described in the response to Question 16.

In the reload fuel safety analysis performed using RETRAN-02, the analysis inputs may be either conservative (license basis) or nominal. The license basis inputs used in the analysis process are identified in WPPSS-FTS-129, Rev. 1. The use of license basis inputs adds a significant amount of conservatism to the analysis process. The nominal inputs include the plant geometry inputs and the kinetic parameters. The plant geometry inputs are calculated based on the "as built" plant configuration. These parameters are subject to only very small uncertainties.

It is the kinetics parameters that are used to establish the overall model uncertainty. The calculation of the model uncertainty is described in Appendix A of the Applications Topical Report Revision 1, and in the response to Questions 7 and 12. The basis for the selection of the magnitude of the uncertainties in the individual parameters making a significant contribution to the overall model uncertainties is also provided in Appendix A of the Applications Topical Report Revision 1.

Question 7

Since the overall uncertainty due to the individual model and input uncertainties was determined by taking the square root of the sum of the squares of the ΔCPR, their statistically independence is inherently assumed. This assumption is justifiable only if coefficients were determined by perturbation analysis. Explain in depth and justify the methodology for determination of these coefficients.

Response

As shown in Tables A-1 and A-2 of the Applications Topical Report Revision 1, the Overall Model Uncertainty due to the individual model and input uncertainties was determined by taking the square root of the sum of the squares (SRSS) of the ΔRCPR due to the uncertainty of each of the relevant model and input parameters. The ΔRCPR attributable to each parameter uncertainty was determined by calculating RCPR with that parameter changed and then comparing the calculated RCPR to the RCPR calculated with that parameter at its nominal best-estimate value. This approach by itself is not a proof that the parameters are statistically independent. However, as explained below, the way the Overall Model Uncertainty is applied to the process of calculating thermal limits (i.e. ΔCPRs) leads to the conclusion that the approach is adequate and the final results are conservative.

The model and input parameter uncertainties that contribute to the Overall Model Uncertainty fall into five groups: Nuclear Model; Core Thermal Hydraulics; Recirculation System; Steam Line Model; and Vessel and Loop Geometry. The parameters chosen, the individual parameter uncertainties to be applied, and the procedure for determining the Overall Model Uncertainty from the individual uncertainties are based on NRC precedents as described in Appendix A of the Applications Topical Report revision 1.

The uncertainties in terms of ΔRCPR for the parameters given in Tables A-1 and A-2 of the Applications Topical Report were combined to obtain the overall model uncertainties using the

SRSS approach. This approach is the same as the one used by a fuel vendor (Reference 1) and other utilities (e.g., References 2 and 3) and has been approved by the NRC. This approach is justified for the following reasons. Where it is more conservative, the bounding values of the uncertainty ranges as recommended by the NRC, instead of the fuel vendor's values, were used (see Table 10-1 in the response to Question 10). Additional parameters not used by General Electric and the NRC were added to the model uncertainty analysis (see the same table). Therefore, the approach used by the Supply System in determination of the model uncertainties is adequate and conservative.

It should be noted that the WNP-2 Licensing Model consistently yields more conservative values in ΔCPR than those of the vendor's (Siemens Power Corporation) as discussed in detail in the Applications Topical Report. The application of the model uncertainty to the Licensing Model is an added conservatism that will make Supply System results even more limiting than SPC's.

REFERENCES

1. NRC Safety Evaluation for the General Electric Topical Report, Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, Volume 1, NEDO-24154-A, August 1986.

2. M.A. Alammar, "The NRC Review of Oyster Creek Reload Licensing Model", Proceedings: Sixth International RETRAN Conference, EPRI NP-6949, August 1990.

3. S.L. Forkner, et al., BWR Transient Analysis Model Utilizing the RETRAN Program, TVA-TR81-01, Tennessee Valley Authority, 1981.

Question 8

Explain in depth and provide thorough justification for the statement that "the contribution of drift flux parameter uncertainty to overall model uncertainty is small (p. A-4)." Provide references for the NRC recommended values for parameters.

Response

Section A.1.2 of the Applications Topical Report Revision 1 provides the discussion of core thermal-hydraulic parameter uncertainty contribution to the overall model uncertainty. For the purpose of uncertainty evaluation of the drift flux model, the NRC staff recommends using a limiting value of 1.0 for the concentration parameter C_o, and a 30% variation in the drift velocity V_{gj}. These NRC staff recommendations are cited in Reference A2 of the Applications Topical Report Revision 1.

| The limiting value of 1.0 for C_o was analyzed by increasing the RETRAN-02 input parameter KAPPA1 by 0.20 (i.e., increasing KAPPA1 from 0.8 to 1.0. However, since KAPPA1=1.0 is a singular point in the drift flux correlation, a value of 0.99 was used for KAPPA1). The 30% variation in Vgj was analyzed by changing the RETRAN-02 input parameter CGL. The ΔRCPR for these parameter variations is shown in Tables A-1 and A-2 of the Applications Topical Report Revision 1 as cases 001/202 and 001/204, respectively. For 8x8 fuel, the C_o variation produced a ΔRCPR of 0.007 and the V_{gj} variation produced a ΔRCPR of 0.000. For 9x9 fuel, the C_o variation produced a ΔRCPR of 0.008 and the V_{gj} variation produced a ΔRCPR of 0.003.

The drift flux parameter uncertainty is included in the overall model uncertainty even though its contribution to the overall model uncertainty is shown to be small to moderate. In fact, all of the uncertainty evaluations shown in Tables A-1 and A-2 were included in the overall model uncertainty regardless of the magnitude of their contribution to the overall model uncertainty.

Question 9

One key parameter missing from parameters considered in the SCU method is an uncertainty associated with the use of the RETRAN code. Considering difficulties experienced with plant data benchmark calculations, justify not including its factor in the overall uncertainty (or include it).

Similarly there is considerable flexibility in the core nodalization used with the VIPRE code. Justify the impact of core nodalization on the overall uncertainty and provide a separate uncertainty associated with it.

Response

As demonstrated in the BWR Transient Analysis Model Topical Report (WPPSS-FTS-129, Rev. 1), the WNP-2 model conservatively predicts all three of the Peach Bottom Unit 2 turbine trip tests when nominal or best estimate input are used for the test conditions. Therefore, use of the RETRAN-02 code and plant model actually introduces a conservative bias.

The process used by the Supply System in determining the overall model uncertainty is consistent with the process used in other applications (See Reference 1 and 2). In this process, the transient analysis code is used to establish the overall model uncertainties through sensitivity studies performed for selected analysis input parameters. It should be noted that for parameters that are not statistically treated in the SCU process, conservative or license basis values are used. This adds a substantial amount of conservatism to the process.

The core nodalization in the Supply System VIPRE-01 models is selected based on results of sensitivity studies. Finer nodalization has insignificant impact on MCPR, as documented in a VIPRE-01 Verification Report submitted to the NRC supporting the VIPRE-01, MOD-02 generic review (Reference 3). Additional justifications for using 25 six-inches axial nodes for the core active region are:

1. To be consistent with interfacing codes - Axial power distributions are calculated in RETRAN-02 and SIMULATE-E for 25 axial nodes. They become part of VIPRE-01 input. This is also consistent with SPC's core nodalization in XCOBRA-T.

2. To be consistent with benchmark analyses - All critical heat flux tests benchmark analyses (steady-state and transient) were performed with a VIPRE-01 model having 25 axial nodes as discussed in the Applications Topical Report Revision 1. Benchmark analyses performed by SPC (Reference 4) were used to determine the ANFB Critical Power Correlation uncertainty. This uncertainty, which includes nodalization uncertainty, is accounted for in the calculation of the safety limit. Therefore, nodalization uncertainty is not included in the evaluation of ΔCPR.

The Supply System will evaluate future plant and fuel design changes and their impact on the applicability of the overall model uncertainty. If a determination is made that the impacts are not negligible, revised analysis of the model uncertainty will be performed and the resulting values will be used in the calculation of the Operating Limit MCPR.

REFERENCES

1. NRC Safety Evaluation for the General Electric Topical Report, Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, Volume 1, NEDO-24154-A, August 1986.

2. S.L. Forkner, et al., BWR Transient Analysis Model Utilizing the RETRAN Program, TVA-TR81-01, Tennessee Valley Authority, 1981.

3. Letter to USNRC from Y.Y. Yung (VMG), Notification of Release and Request for NRC Review of VIPRE-01 MOD-02, February 28, 1990.

4. ANFB Critical Power Correlation, ANF-1125(P)(A), ANF-1125 Supplement 1(P)(A) and

ANF-1125 Supplement 2(P)(A), April 4, 1990.

Question 10

The Supply System extrapolated the findings obtained by GE for its GESSAR plant using its own code with respect to the degree of sensitivity exhibited by certain parameters to the expected values when RETRAN and VIPRE are used in the similar analysis. Justify in depth the assumption of applicability of GE results to the WNP-2 RETRAN/VIPRE analysis. Discussion should include identification of the similarities and differences of code models used to model relevant components between the codes. If the Supply System is unable to do the foregoing, it must otherwise justify using the same values for the range of uncertainties and variations in the analysis and identify biases due to these codes.

Response

The Supply System has not extrapolated the findings obtained by GE for its GESSAR plant using its own code with respect to the degree of sensitivity exhibited by certain parameters to the expected values when RETRAN and VIPRE are used in the similar analysis. Rather, the parameters and their uncertainties used in the determination of overall model uncertainty were determined based on NRC precedents as described in Appendix A of the Applications Topical Report Revision 1. The response to Questions 6 and 7 provides further information.

The parameters and uncertainties used by others in determination of overall model uncertainty are shown in Table 10-1. The GE/ODYN and NRC/ODYN parameters are discussed in Reference 1 (same as Reference A2 of the Applications Topical Report Revision 1). The TVA parameters are discussed in Reference 2 (same as Reference A4 of the Applications Topical Report Revision 1). Note that the TVA analysis used RETRAN as the system transient computer code. As can be seen from Table 10-1, the Supply System approach is consistent with that of other BWR models and is not simply an extrapolation of GE findings. The bases for the uncertainties assigned to each of the parameters considered in the overall model uncertainty is provided in Appendix A of the Applications Topical Report Revision 1.

Further discussions on the justification of the selection of the parameter uncertainties are given in the response to Question 4. It should be emphasized that even though the GE study quoted here (Reference 1) and in the response to Question 4 were for the qualification of GE's ODYN code, the process for the derivations of the uncertainties of individual parameters is independent of the codes used (see References 2 and 3 in Question 4). The only linkage of the uncertainties with any code is through the calculation of the ΔRCPRs using these uncertainties. Since the ΔRCPRs in the Supply System methodology are calculated through a RETRAN model and the model uncertainty is obtained by combining these ΔRCPRs, it will be applicable to transient analysis using the RETRAN code.

REFERENCES

1. NRC Safety Evaluation for the General Electric Topical Report, Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, Volume 1, NEDO-24154-A, August 1986.

2. S.L. Forkner, et al., BWR Transient Analysis Model Utilizing the RETRAN Program, TVA-TR81-01, Tennessee Valley Authority, 1981.

Table 10-1
Model Uncertainty Parameters

SUPPLY SYSTEM	TVA	GE/ODYN	NRC/ODYN
NUCLEAR MODEL PARAMETERS			
Void Coefficient (13%)	Void Coefficient (13%)	Void Coefficient (13%)	Void Coefficient (11%)
Doppler Coefficient (-10%)	Doppler Coefficient (-10%)	Doppler Coefficient (-6%)	Doppler Coefficient (-10%)
Prompt Moderator Heat (-25%)	Prompt Moderator Heat (-25%)	Prompt Moderator Heat	Prompt Moderator Heat
Scram Reactivity (-10%)	Scram Reactivity (-10%)	Scram Reactivity (-4%)	Scram Reactivity (-10%)
CORE THERMAL-HYDRAULIC PARAMETERS			
Correlation (Kappa1+0.20)	Core Press Drop (+1.5psi)	Drift Flux (Co 3%;Vgj 20%)	Drift Flux (Co=1;Vgj 30%)
Correlation (CGL+30%)	Core Loss Coeff (5% exit)	Subcool Void Model (n=1.25)	Subcool Void Model (n=.5, 2)
Correlation (CDB+20%)	Core Noding (increase to 24)	Core Press Drop (+1.5psi)	Core Press Drop (+1.5psi)
Correlation (CHN+20%)	Core Bypass Flow (-20%)		
Core Loss Coeff (-20%)	Th-Hyd Method (Use HEM)		
Fuel Pin Noding (+50%)	Subcooled Voids (-30%)		
Core Power (+4%)			
RECIRCULATION SYSTEM PARAMETERS			
Recirulation Loop Inertia (+100%)	Recirculation Loop Inertia (+100%)	System Inertia (+200%)	System Inertia (+200%)
Recirculation Pump Head (-10%)	Recirculation Pump Head (-10%)	Jet Pump Losses (-20%)	Jet Pump Losses (-20%)
Jet Pump Inertia (+100%)	Jet Pump Inertia (+100%)	Separator Inlet Inertia (-30%)	Separator Inlet Inertia (2x)*
Separator Liquid Outlet Inertia (+100%)	Jet Pump M Ratio (+7%)		
Jet Pump Loss Coeff (-20%)	Jet Pump Head (+10%)		
Separator Inlet Inertia (-30%)	Separator Liquid Outlet Inertia (+100%)		
	Vessel Inertia		
MAIN STEAM LINE MODEL PARAMETERS			
Steam Line Inertia (+7%)	Steam Line Inertia (+7%)	Specific Heat Ratio	Specific Heat Ratio
Steam Line Loss Coeff (-20%)	Steam Line Press Drop (-10%)	Steam Line Loss Coeff (-20%)	Steam Line Loss Coeff (-20%)
VESSEL AND LOOP GEOMETRY PARAMETERS			
Steam Dome Volume (-5%)	Steam Dome Volume (-5%)	None Considered	None Considered
Steam Line Volume (-5%)	Steam Line Volume (-5%)		
	Upper Downcomer Volume (-5%)		
	Steam Line Area (-5%)		

*NRC recommends a factor of two in Separator Inlet Inertia but shows -200% in Table I of Reference 1. Note that -200% would produce a change in sign for the Separator Inlet Inertia

Question 11

The Supply System has elected to use algebraic slip for modeling of the phases present in the vessel with RETRAN and it is also using the drift flux model with VIPRE. Neither of these models has been qualified. As per SER on WPPSS-FTS-129, Rev. 1, qualify both of these models for use with WNP-2 analysis. Discussion should include applicability and scalability of qualification provided by the Supply System (based upon the one small scale experiment which it used) to the full-scale plant.

Response

The EPRI algebraic slip and drift flux models have been qualified for use in RETRAN-02 for BWR Transient and Application Topical Analysis Reports. At the time the initial SER for RETRAN-02 was released, the validation and qualification of these models had not been completely documented. After the RETRAN-02 review began, a report (Reference 1) documenting the models has been made available and these models have been used by many U.S. utilities for Transient and Applications Topical Reports using RETRAN-02 as well as for individual reload licensing analyses and submittals to the NRC. The EPRI drift flux model in VIPRE-01 MOD02 is the same as the model in RETRAN-02 and as documented in Reference 1.

The configurations used in Reference 1 for model validation included bundle, tube, and channel geometry and the experiments were performed for a wide range of pressure, flow, and heating conditions. The geometry of the FRIGG BWR tests was quite similar to that found in commercial BWR core designs.

As shown in the BWR Transient Analysis Model Topical Report WPPSS-FTS-129, Rev. 1, the use of the algebraic slip model has been demonstrated to accurately predict the response of the Peach Bottom Unit 2 turbine trip tests. Uncertainty associated with the parameters in the algebraic slip model has been considered in determination of the overall model uncertainty (See

the response to Questions 6 and 7). The contribution of this uncertainty to the overall model uncertainty is small.

Additional justification of the drift flux model in VIPRE-01 for use with WNP-2 analysis is provided in the response to Question 1. The drift flux model, in combination with other flow and heat transfer correlations specified in the Supply System VIPRE-01 models, was demonstrated to yield conservative results for BWR CPR analyses and good agreements of benchmark analysis results with measured data, as discussed in the response to Question 1.

REFERENCE

1. G.S. Lellouche and B.A. Zolotar, Mechanistic Model for Predicting Two-Phase Void Fraction for Water in Vertical Tubes, Channels and Rod Bundles, EPRI NP-2246-SR, February 1982.

Question 12

Explain how the physics, systems, and modeling uncertainties are statistically treated as part of the overall Response Surface Method (RSM).

Response

Statistical combination of uncertainty (SCU) methodology is used as a part of the WNP-2 reactor analysis methodology only when the deterministic (license basis) approach can lead to technical specification setpoints or core operating limits that can result in undesirable plant operating restrictions. The SCU methodology defines a set of more operationally acceptable limiting conditions while retaining an appropriate level of conservatism. In the SCU methodology, selected analysis input parameters are assumed to be at their nominal or best estimate values, and their uncertainties are statistically combined to establish the amount of conservatism to be applied to the results. The analysis parameters that are not treated statistically in the analysis process are input at their license basis values. This treatment of input parameters adds a substantial amount of conservatism to the SCU process. For example, if the control rod scram insertion time were the only parameter treated statistically, then sufficient event analyses would be performed varying the scram time to obtain an accurate statistical statement about the probability of the result. In this example, all other input parameters would be the same as for the license basis analysis process.

The statistical simulation of an event requires that a large number of cases be run to obtain the desired accuracy and reduce the uncertainty associated with the analysis results. The response surface methodology provides an efficient way of reducing the number of cases required to make an accurate statistical statement about the event analysis results. The response surface is an algorithm that approximates the event analysis codes results, for the figure of merit over the range of event probabilities as a function of the input parameters included in the development of the response surface. In the WNP-2 reactor analysis methodology, the response surface is constructed using a least squares fit to data generated for specific points of an experimental

design having specific probabilities of occurrence for the independent parameters. A quadratic fit is typically used, although higher level fits could be used if desired. The STARS code (see the response to Question 3) is used to determine the fitting coefficients for the response surface algorithm. As a part of the process to develop the response surface algorithm, a fitting error is calculated that represents the accuracy of the fit and is used in the statistical analysis process.

For the case of a response surface developed for the statistical treatment of control rod scram insertion time for the limiting anticipated operational occurrence (a rapid pressurization event), the response surface is used to predict the ratio of the critical power ratio (RCPR) as a function of scram time. The particular event analyses, that may be performed for a particular experimental design, may include the nominal scram time, ± 1 standard deviation, and ± 2 standard deviations. It should be noted that, for different response surface experimental designs or for more independent parameters, a different number of points or different analysis values may be used to obtain an appropriate level of accuracy for the response surface algorithm. The number of points typically used in the development of the response surface is described in more detail in the response to Question 5.

Once the response surface has been developed, the statistical analysis process can proceed. The statistical analysis consists of performing a Monte Carlo sampling of the response surface using the uncertainty distributions for the input parameters. Also included in the statistical analysis is a Monte Carlo sampling of the fitting error, which is added to the result from the sampling of the response surface. This part of the analysis results in the development of a probability distribution function for the figure of merit for the event. The convolution process for model uncertainties in the STARS code, using the uncertainty distribution, is then used to statistically combine the model uncertainty with the results of the Monte Carlo analysis to establish the overall event analysis probability distribution function. The process for statistically combining the model uncertainties effectively flattens the probability distribution function, which increases the "tails" of the distribution. Because the goal is to demonstrate that there is an acceptably low probability of exceeding the event acceptance limit, increasing the tail of the distribution adds conservatism to the statistical analysis process.

For the previous scram insertion time example, the Monte Carlo analysis typically consists of a sampling of 100,000 points of the response surface and combining each point with a statistical sampling of the fitting error. The STARS convolution process is then used to combine the Monte Carlo distribution with the model uncertainty. The model uncertainties include the nuclear model uncertainties, core thermal hydraulic uncertainties, recirculation system parameters, steam line model parameters, and vessel and loop geometry parameters quantified in Appendix A of the Applications Topical Report Revision 1. As discussed in Appendix A, the model uncertainties are developed using sensitivity studies performed for the key analysis input parameters. In the sensitivity studies, the parameters are varied by a value assessed to be at least two standard deviations. The calculated uncertainties for each parameter are then combined using the square root of the sum of the squares process to obtain the overall model uncertainty. This is the model uncertainty used in the convolution process. The evaluated model uncertainty is input in the statistical analysis process as a normal distribution with standard deviation equal to one-half of the overall model uncertainty.

Question 13

Identify the impact to its hot channel methodology in the mixed core environment.

Response

The hot channel methodology remains unchanged in the mixed core environment. The impacts on thermal margin evaluation due to mixed core are:

1. The procedure to calculate ΔCPR with VIPRE-01 (See Appendix B, Section B.1 of the Application Topical Report Revision 1) is applied to each fuel design assembly (modeled in VIPRE-01 with assembly specific geometry, F-effective and hydraulic data) in the mixed core.

2. The boundary conditions (inputed to VIPRE-01) determined in the RETRAN-02 transient analysis are the results of a simulation that includes the modeling of the mixed core (See the response to Question 16). The boundary conditions are assumed to be common to the individual fuel assemblies of the mixed core, consistent with SPC's BWR transient methodology (Reference 1).

Using the Cycle 8 (a mixed core with 576 SPC 8x8 fuel bundles and 188 SPC 9x9-9X fuel bundles) as an example, two VIPRE-01 hot channel models are developed, one for SPC 8x8 fuel and one for SPC 9x9-9X fuel. These two models have identical input except for the geometric data, loss coefficients and F-effective. Applying the boundary conditions calculated in RETRAN-02 to the SPC 8x8 hot channel model and the SPC 9x9-9X hot channel model respectively as described in the hot channel methodology will provide two ΔCPRs, one for SPC 8x8 fuel and one for SPC 9x9-9X fuel.

In summary, the hot channel methodology uses the forcing function generated by RETRAN-02 and applies it to each fuel type in the core. The forcing function from RETRAN-02 calculation

has imbedded in it the mixed core effect through the mixed core modeling as discussed in the response to Question 16.

REFERENCE

1. XN-NF-80-19(P)(A), Volume 3, Revision 2, EXXON Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description, January 12, 1987.

Question 14

The ANFB critical power correlation developed by Advanced Nuclear Fuels Corporation (ANF) was implemented in the VIPRE-01 code for the thermal margin evaluation. Thorough justification of use of this correlation in the mixed core environment must also be provided.

Response

The ANFB critical power correlation was reviewed and approved by the NRC for CPR calculations (Reference 1). The correlation was developed as a generic tool for evaluating critical power and assessing thermal margin for all SPC BWR fuel designs. Differences between specific fuel designs are accounted for by a lattice position dependent function termed F-effective. The correlation uses assembly averaged values of coolant flow, enthalpy, and pressure, including the factor called F-effective, to predict the bundle average critical heat flux. The corresponding assembly critical power is determined iteratively and equals the assembly power when the predicted ANFB critical heat flux and the bundle average heat flux are equal.

The hydraulic and thermal conditions over which the ANFB is applicable bounds the range of parameters over which the plant transients are performed (Reference 1). The mixed core will remain within the ANFB correlation range of applicability. Therefore the ANFB correlation can be applied to any SPC fuel design bundle, regardless of the mixed core environment.

REFERENCE

1. ANFB Critical Power Correlation, ANF-1125(P)(A), ANF-1125 Supplement 1(P)(A) and ANF-1125 Supplement 2(P)(A), April 4, 1990.

Question 15

Justify the use of the steam line length of 400 ft in the NRC specified problem as a benchmark analysis. The Supply System acknowledged the significance of this value in the computed results but did not state whether the value used was consistent with that used by GE and BNL or an accurate indication of the real plant data.

Response

The actual steam line length for Peach Bottom Unit 2 is 462 ft (Reference 1). In the NRC Specified Licensing Analysis, it was reduced to 400 ft to be consistent with the value used by GE (Reference 2). The length and volume of the steam line have a significant effect on the timing and magnitude of the pressure wave. Therefore consistency is essential for meaningful comparisons.

REFERENCE

1. K. Hornyik and J.A. Naser, RETRAN Analysis of the Turbine Trip Tests at Peach Bottom Atomic Power Station Unit 2 at the End of Cycle 2, EPRI NP-1076-SR, Electric Power Research Institute, April 1979.

2. NRC Safety Evaluation for the General Electric Topical Report, Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154 and NEDO-24154-P, Volumes I, II, and III, June 1980.

Question 16

Although the Supply System mentioned in the cover letter that some changes were made to incorporate the mixed core effects, no details of such incorporation were given. Please provide a through discussion of how these effects are being incorporated in the transient reload analysis, input parameters, MCPR analysis or applicable CPR correlations.

Response

The methodology for the transient analysis and MCPR analysis, as well as the ANFB CPR correlation does not vary due to mixed core environment. The introduction of the mixed core impacts the reload analyses through appropriate changes in input parameters of the computer codes used in the analyses. They are discussed as follows:

1. RETRAN-02 Input

   With the exception of the loss coefficients, all of the fuel related parameters are computed as either sum of all bundles or average of all bundles.

   a. Sum of all bundles.
      For example, the Cycle 8 core consists of 576 8x8 bundle and 188 9x9 bundle.

      Core volume flow area (A) = 576 * flow area of 8x8 bundle
      + 188 * flow area of 9x9 bundle

   b. Average of all bundles.
      For example,
      Clad thickness = (576 * clad thickness of 8x8 bundle
      + 188 * clad thickness of 9x9 bundle)/764

The loss coefficients for the mixed core average bundle are treated as a special case and are determined with the following averaging method.

The pressure losses due to local effects are given as

$$\Delta P = K * (W/A)**2 / (2 * g_c * \rho_l)$$

So,

$$W = sqrt (\Delta P/K) * A * C \quad \text{[where C is a constant]}$$

Assuming the multichannel pressures are allowed to equalize at each elevation,

$$W = \sum w_i = \sum sqrt (\Delta P/K_i) * A_i * C$$

$$sqrt (1/K) * A = \sum sqrt (1/K_i) * A_i$$

where i specifies individual bundle.

So, for the Cycle 8 mixed core:

$$sqrt (1/K) = sqrt (1/K_{8x8}) * A_{8x8}/A * 576$$
$$+ sqrt (1/K_{9x9}) * A_{9x9}/A * 188$$

where $A = 576 * A_{8x8} + 188 * A_{9x9}$

and K is the mixed core equivalent loss coefficient.

2. SIMULATE-E & VIPRE-01 Inputs

These two codes model either the individual bundle or the bundles of the same fuel design lumped together. Data for different fuel design bundles (geometry, hydraulic characteristics) are specified separately. Mixed core modeling simply requires the entering of additional data, one set of data for each fuel bundle design.

Question 17

Identify and justify any changes to the input (initial conditions, transient assumptions, trip setpoints and delays) for these transients from those assumed in the current FSAR transients.

Response

The input parameters for licensing analysis are based on Table 15.0-2 "Input Parameters and Initial Conditions for Transients" in the current WNP-2 FSAR. Since the RETRAN-02 model requires more data than those listed in the FSAR table, some input data were obtained from GE Design Specification Documents. These parameters are listed in Table 4.1 of the Transient Topical Report (Reference 1).

The parameters that are not given in the FSAR but were obtained from GE Design Specifications are:

1. Safety/Relief Valves (SRV) Relief Function Opening Delay Time

   A licensing value of 0.4 seconds was used. GE Design Specifications for the Nuclear Boiler System for WNP-2 (Reference 2) requires a value of ≤ 0.1 seconds. Thus, the use of 0.4 seconds is conservative for the analysis of thermal limits.

2. Low Water Level (Level 3) Scram Setpoint

   WNP-2 Technical Specifications Setpoint for low water level scram is 13 inches above instrument zero. For licensing analysis, a value of 7.5 inches was used. This conservative value is the "analytical" value given in the GE Design Specifications Document for Nuclear Boiler System for WNP-2 (Reference 2).

3. Turbine Stop Valve (TSV) Closure Position Scram

WNP-2 Technical Specifications Setpoint for TSV closure position scram is 5% closed. Based on GE Design Specifications (Reference 3), an "analytical value" of 10% was used in the RETRAN-02 model.

4. Main Steam Line Isolation Valve (MSIV) Closure Position Scram

WNP-2 Technical Specifications Setpoint for MSIV closure position scram is 10% closed. Based on GE Design Specifications (Reference 2), an "analytical value" of 15% was used in the RETRAN model.

It should be noted that the conservatism of the recirculation pump moment of inertia (see Table 4.1 of the Transient Topical Report, Reference 1) is removed for the transient analyses as presented in the Applications Topical Report Revision 1. This change is justified by the fact that the pump moment of inertia is a well-defined parameter. Use of nominal value is consistent with current industry-wide practice.

REFERENCES

1. Y.Y. Yung, et al., BWR Transient Analysis Model, WPPSS-FTS-129, Rev. 1, Washington Public Power Supply System, September 1990

2. General Electric Design Spec Data Sheet, Nuclear Boiler System, 23A1886AA, Rev. 12, January 1989

3. General Electric Design Spec Data Sheet, Reactor Protection System, 23A1877AA Rev.2, March 1986

Question 18

On the Basis of DCPR comparison with the vendor's results the Supply System predictions appear to be conservative. However, in order to ensure that the comparison is valid, provide comparative system parameter plots for representative parameters for each transient. Where there are significant differences between the results of those two sets of calculations, provide detailed discussion and explanation of the source(s) of differences.

Response

System response calculations for the three primary transients: a) Load Rejection without Bypass (LRNB), b) Feedwater Controller Failure (FWCF), c) Main Steam Isolation Valves Closure (MSIVC), were presented in the Application Topical Report. The comparative system parameter plots (Supply System Vs. SPC) are provided here with explanation of the differences.

LRNB

Figures 18.1 through 18.7 show comparisons of the results from RETRAN-02 and from SPC's licensing analysis for the representative parameters for the Load Rejection without Bypass (LRNB) Transient. The SPC results were obtained from Reference 1 and are based on SPC's transient code COTRANSA2. Figure 18.1 indicates that the power excursion as predicted by RETRAN-02 matches that of COTRANSA2 reasonably well. The RETRAN-02 result yields a higher peak power than COTRANSA2. This higher peak power is caused mainly by the higher pressurization rate (discussed in detail below).

Figure 18.2 shows the core average heat flux versus time. This plot shows the integrated effect of the core power history of Figure 18.1. As a result of higher core power, the Supply System result is more conservative than SPC's. The heat flux has a direct impact on the ΔCPR calculation. Generally, conservative heat fluxes yield conservative ΔCPR's.

Figure 18.3 shows the core inlet flow versus time. The vendor's result does not show as large an initial increase in core flow as RETRAN-02 predicts. This difference is related to the different way of modeling the separator and upper downcomer regions. SPC models the separator and upper downcomer as one volume whereas Supply System's RETRAN-02 model models the two regions as two volumes. During a pressurization transient such as LRNB, the pressure wave in SPC's model would reach the core region at the same time as it would reach the upper downcomer region whereas the pressure wave in the RETRAN-02 model would take longer to reach the core region than the upper downcomer region, thus retarding the flow increase caused by the pressure wave effect through the downcomer and lower plenum. However, as one can see, the timing of flow changes match each other reasonably well indicating that the pressure wave predictions are close to each other.

Figure 18.4 gives a comparison of dome pressures. As discussed above, the COTRANSA2 pressurization rate is slower than that of RETRAN-02. The different rate of pressurization is largely caused by the hydrodynamic modeling differences of the two models. In COTRANSA2, the separator-upper downcomer region is modeled using the equilibrium modeling approach, where the steam and liquid are assumed to have the same temperature at all times. In the RETRAN-02 model, the upper downcomer region (which is modeled separately from the separator) is modeled by the "non-equilibrium" model, where the steam and liquid temperatures are calculated separately. As the pressure wave travels from the turbine control valves to the steam dome and upper downcomer, the compressed steam has a higher temperature than the subcooled liquid in the upper downcomer, the pressure will be kept higher than would be predicted with an equilibrium model. Therefore, the RETRAN-02 model with "non-equilibrium" upper downcomer yields a higher pressurization rate, thus a conservative (larger) power increase. It should be noted that the SPC calculation does not show a turnaround of the pressure because their simulation was terminated at 3 seconds. Turnaround of the dome pressure further into the transient is expected because of the operation of safety/relief valves.

Figure 18.5 gives the steam flow versus time at the junction of steam line and pressure vessel. As indicated, the pressure wave predictions of RETRAN-02 and COTRANSA2 are very close.

Figure 18.6 shows the water level comparison. The COTRANSA2 prediction does not show as profound an initial drop as RETRAN-02 predicted. This diverged behavior is due to the combined effects of modeling differences in the separator and upper downcomer region and the equilibrium assumptions. As mentioned above, SPC models the separator and upper downcomer as one volume and it uses the equilibrium model. This difference in modeling will cause a difference in the depression of the water level as the pressure wave reaches the upper downcomer region.

Figure 18.7 gives a comparison of the feedwater flow. RETRAN-02 model allows the feedwater control system to control the flow during the transient whereas SPC's model assumes a constant feedwater flow throughout the transient. The difference in the feedwater flow assumption does not have a significant impact on the analysis results, given the fact that the minimum CPR typically occurs in about one second after initiation of the LRNB transient.

<u>MSIVC</u>

See the response to Question 28

<u>FWCF</u>

The comparative system parameter plots for FWCF transient are presented in Figures 18-8 through 18-21. As can be seen, all representative parameters compared well with SPC's results except the feedwater flow and water level. The sources of the differences, in addition to the modeling differences discussed above for the LRNB transient, are:

1. The Supply System analysis conservatively increases the feedwater flow with a step change and the SPC analysis uses a time constant to control the rate of the feedwater flow increase. Two cases were run with a ten second linear ramp feedwater flow increase to investigate the sensitivity of the transient response to the rate of feedwater flow increase. This analysis showed that the step increase is slightly more conservative. With

the linear ramp increase, there is no change in ΔCPR for 104%P/106%F case and a decrease of 0.01 in ΔCPR for 47%P/106%F case.

2. In the Supply System analysis, feedwater pump trip occurred when the reactor vessel water level reached high water level (L8) setpoint. Feedwater flow then coasted down to zero in about 5 seconds. In the SPC analysis, feedwater pump trip was ignored and feedwater flow remain constant at its maximum run out capacity. This modeling difference has an insignificant effect on the transient response since scram followed shortly (activated by TSV closure when water level reached L8) and core power and heat flux decreased rapidly. A case was run with no feedwater pump trip and confirmed that there is no change in ΔCPR. The Supply System modeling of feedwater flow is consistent with GE's approach as presented in WNP-2 FSAR Chapter 15.

For a more exact comparison on the differences in water level predictions for FWCF transients, calculations were made by forcing the feedwater flow increase to be identical to the vendor's. The resulting water levels were compared in Figure 23 of Reference 2 and discussed in response to Question 6 in the same reference. It showed that RETRAN predicted a lower water level throughout the FWCF transient, consistent with the LRNB predictions in Figure 18-6.

As discussed in response to Question 6 in Reference 2, the LRNB transient is not impacted by the water level predictions. In the same reference, additional analysis for the FWCF showed that the lower water level leads to a delay in the turbine trip and time to scram. These delays resulted in an increase in peak heat flux from 120.2%NBR to 121.4%NBR, making the RETRAN water level calculation more conservative in determination of thermal limits.

It should be noted that, even though in Figure 18-13 the RETRAN model showed slightly higher water level at the point of turbine trip for the 47% power case for FWCF, the governing case for the FWCF transient is at full power. Figure 18-20 for the full power case indicates that the water levels at the time of turbine trip is essentially the same for both RETRAN and vendor's calculation (considering the fact that RETRAN did not use a ramped feedwater flow run-up).

Given the same water level, the Supply System methodology still resulted in larger ΔCPRs (see Section 5.3.7 of the Applications Topical) assuring the model's conservatism. It should be noted that Figures 18-13 and 18-20 showed a diverged comparison in water levels after Level 8 is reached. This is due to the modeling differences as discussed in Item 2 on the previous page. As stated, this has no impact on the CPR calculations.

REFERENCE

1. J.G. Ingham, WNP-2 Cycle 8 Plant Transient Analysis, EMF-92-039, Rev. 1, Siemens Power Corp., Richland, Washington, June 1992

2. Letter from G.C. Sorensen (Supply System) to NRC, Nuclear Plant No.2, Operating license NPF-21 Response to Second Request for Additional Information Regarding topical Report WPPSS-FTS-129, "BWR Transient Analysis Model" (TAC No. 77048), March 5, 1992.

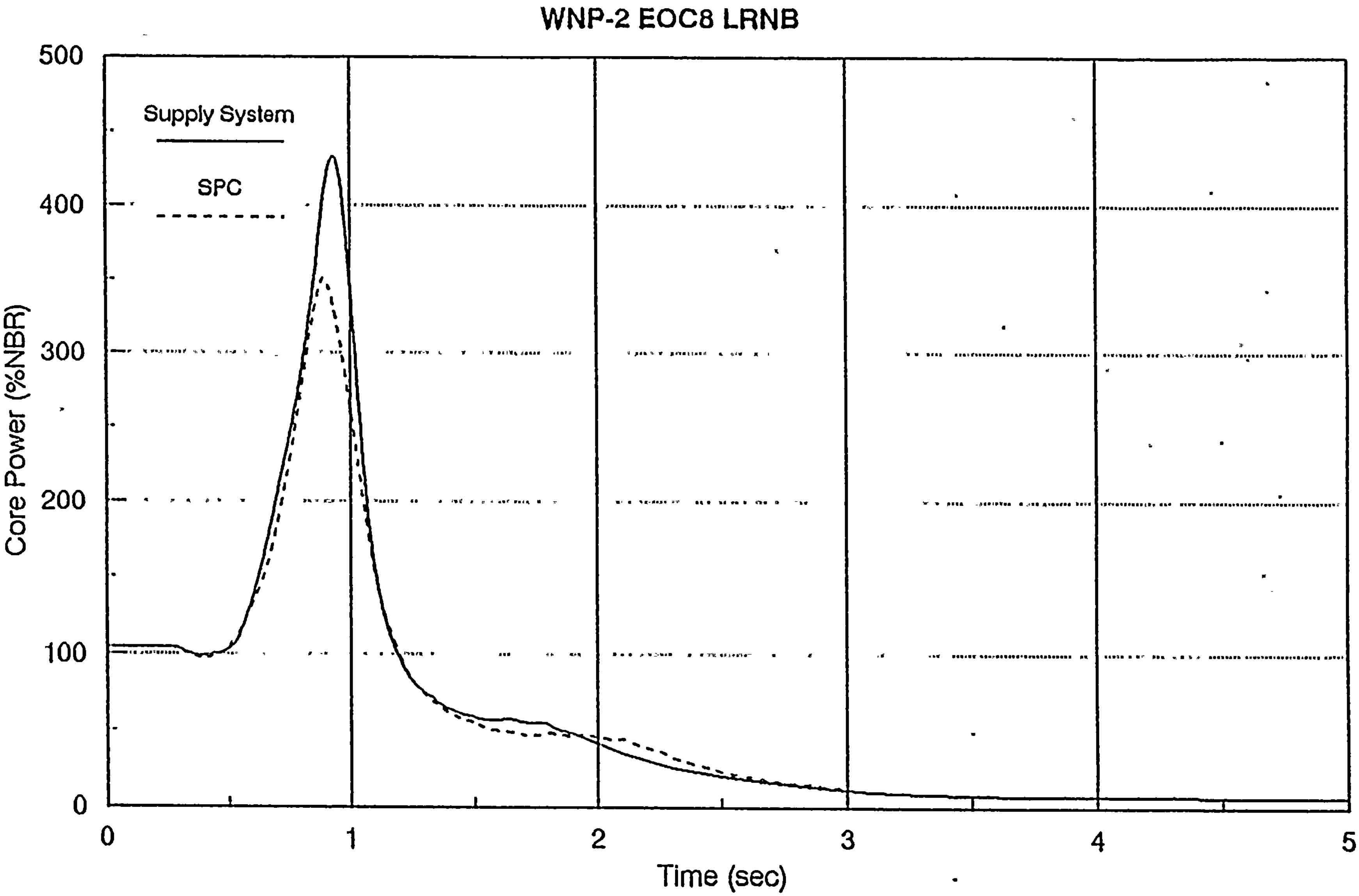


Figure 18-1 LRNB Results, RPT Operable,
Tech. Spec. Scram Time

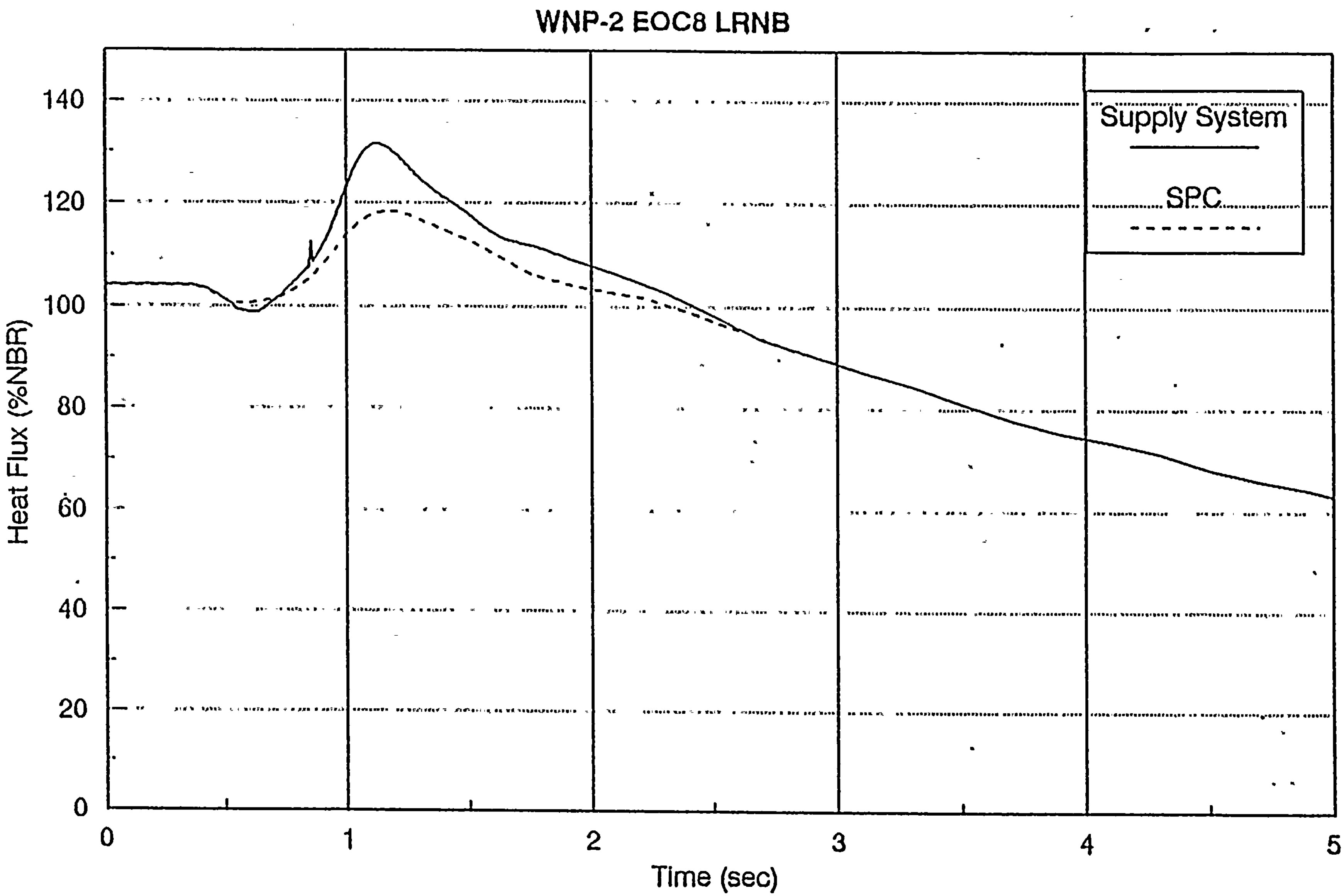


**Figure 18-2 LRNB Results, RPT Operable,
Tech. Spec. Scram Time**

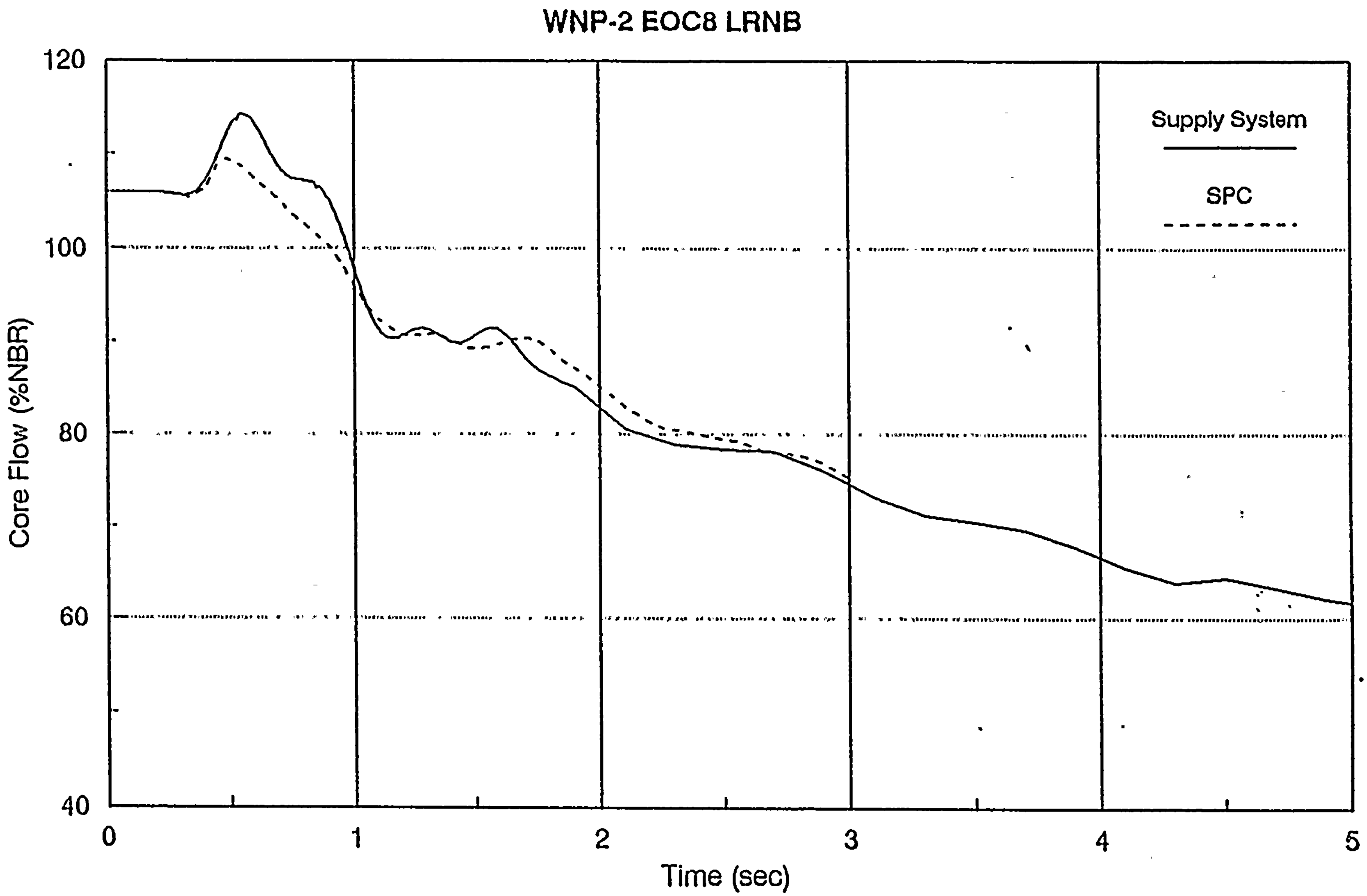


Figure 18-3 LRNB Results, RPT Operable, Tech. Spec. Scram Time

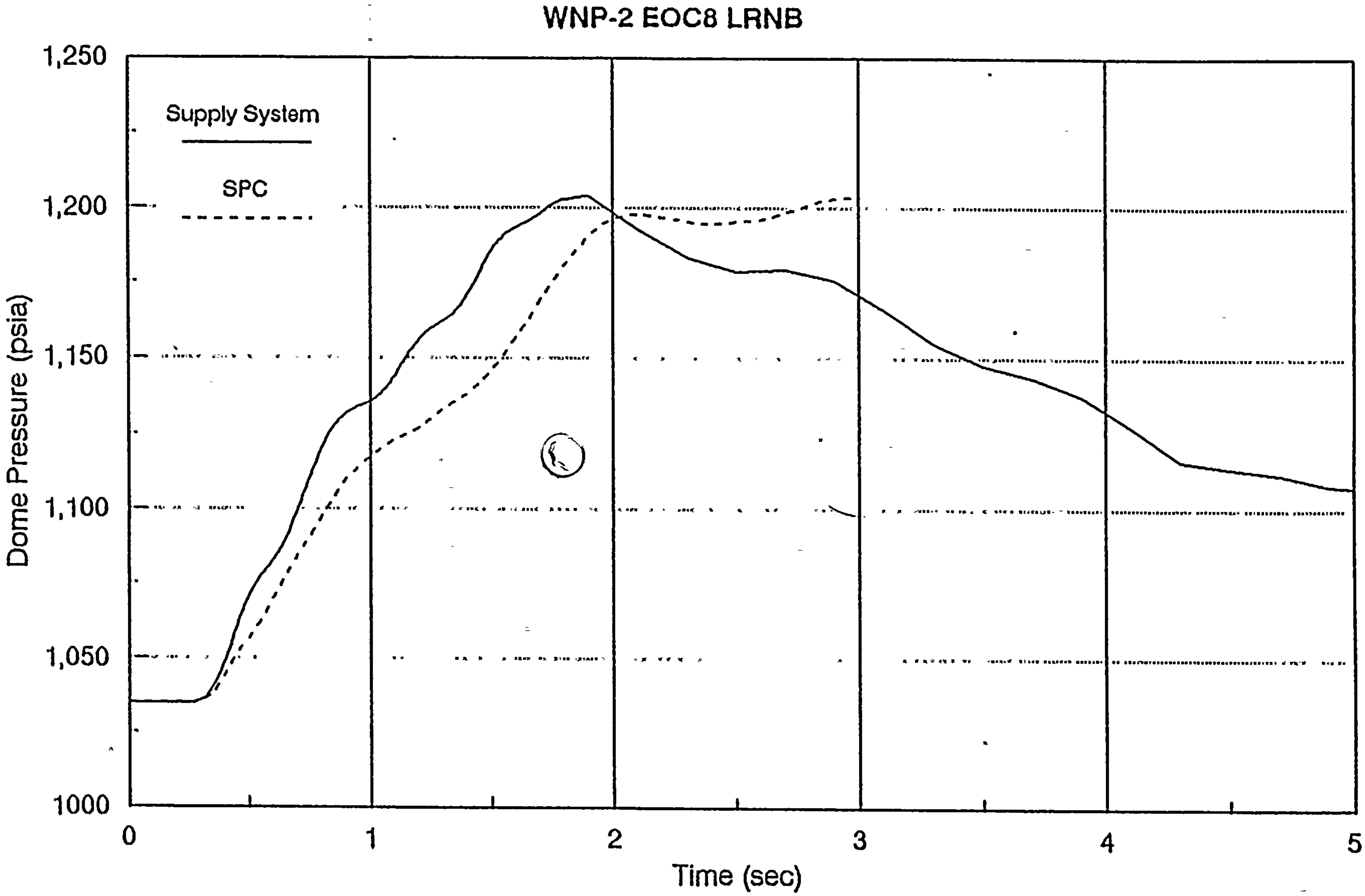


Figure 18-4 LRNB Results, RPT Operable, Tech. Spec. Scram Time

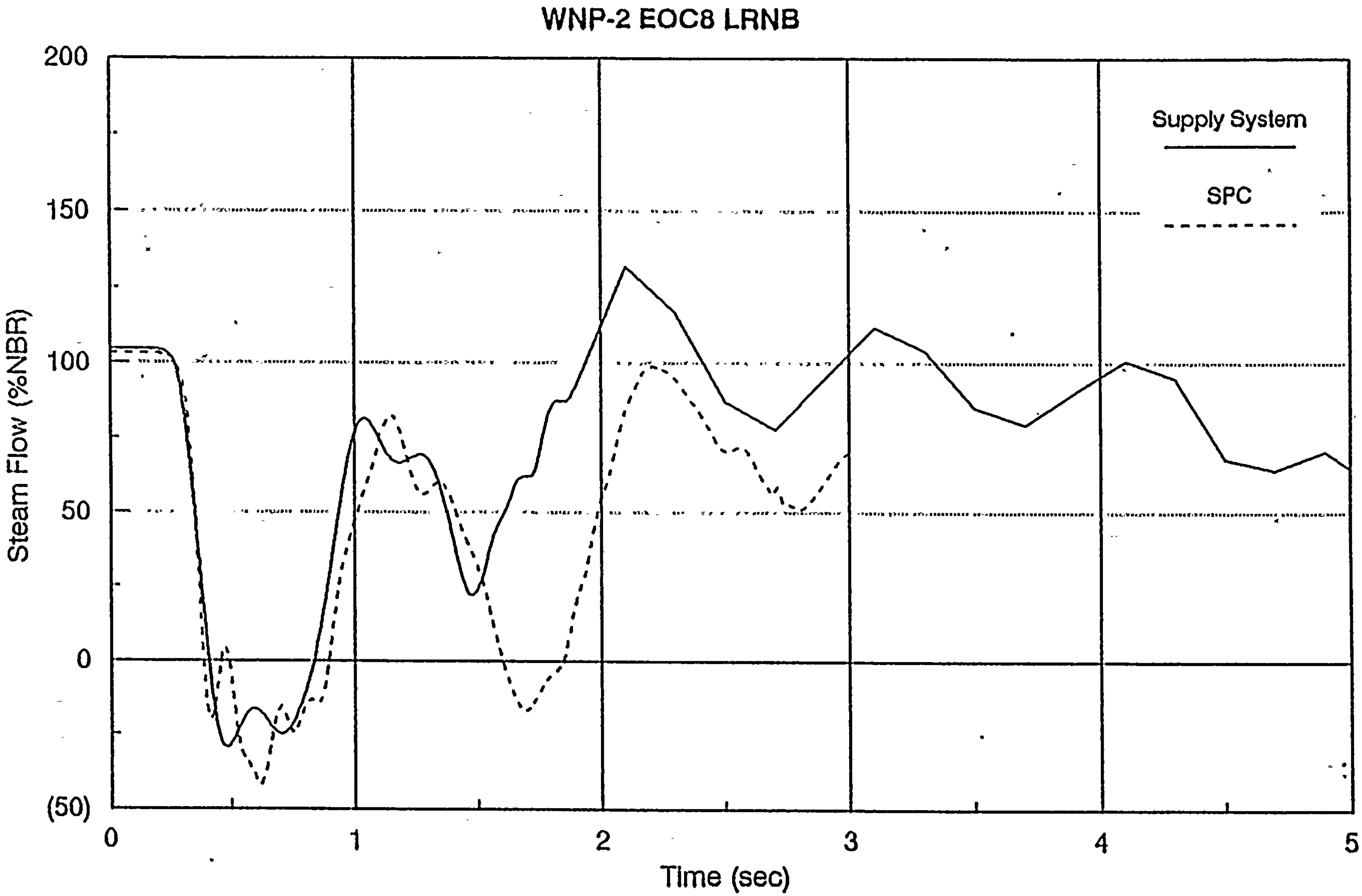


Figure 18-5 LRNB Results, RPT Operable,
Tech. Spec. Scram Time

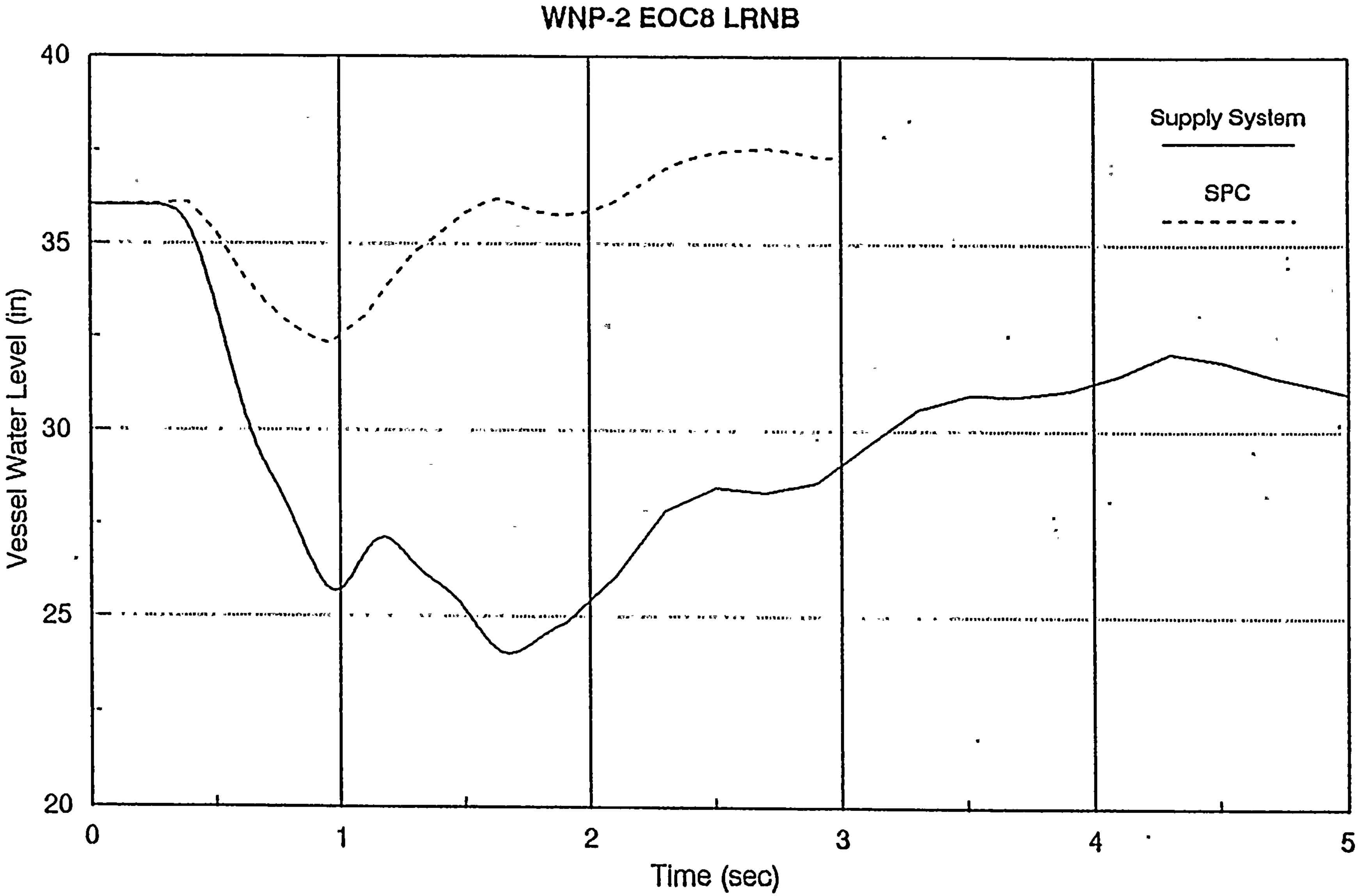


Figure 18-6 LRNB Results, RPT Operable, Tech. Spec. Scram Time

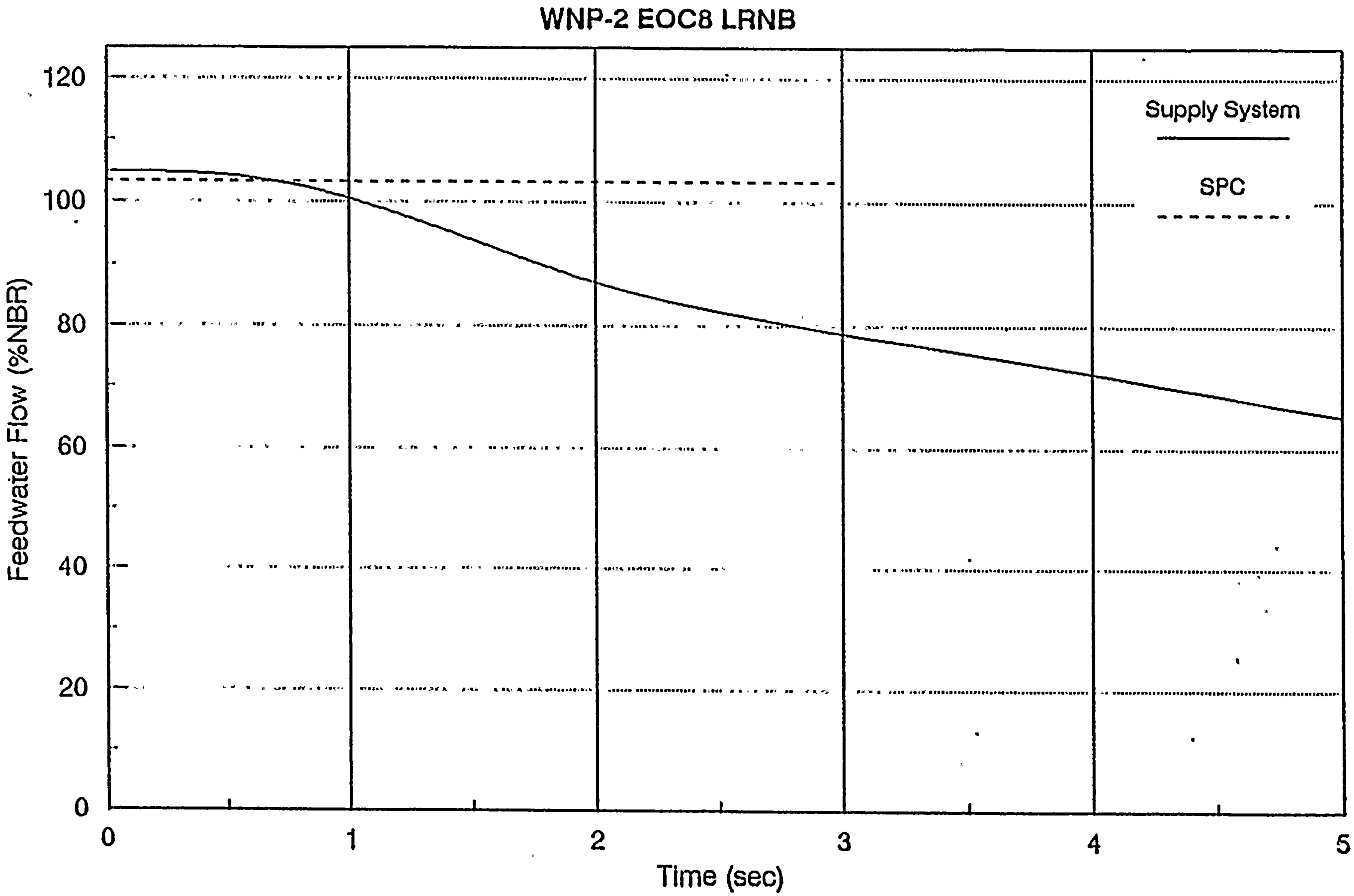


Figure 18-7 LRNB Results, RPT Operable,
Tech. Spec. Scram Time

WNP-2 EOC8 FWCF

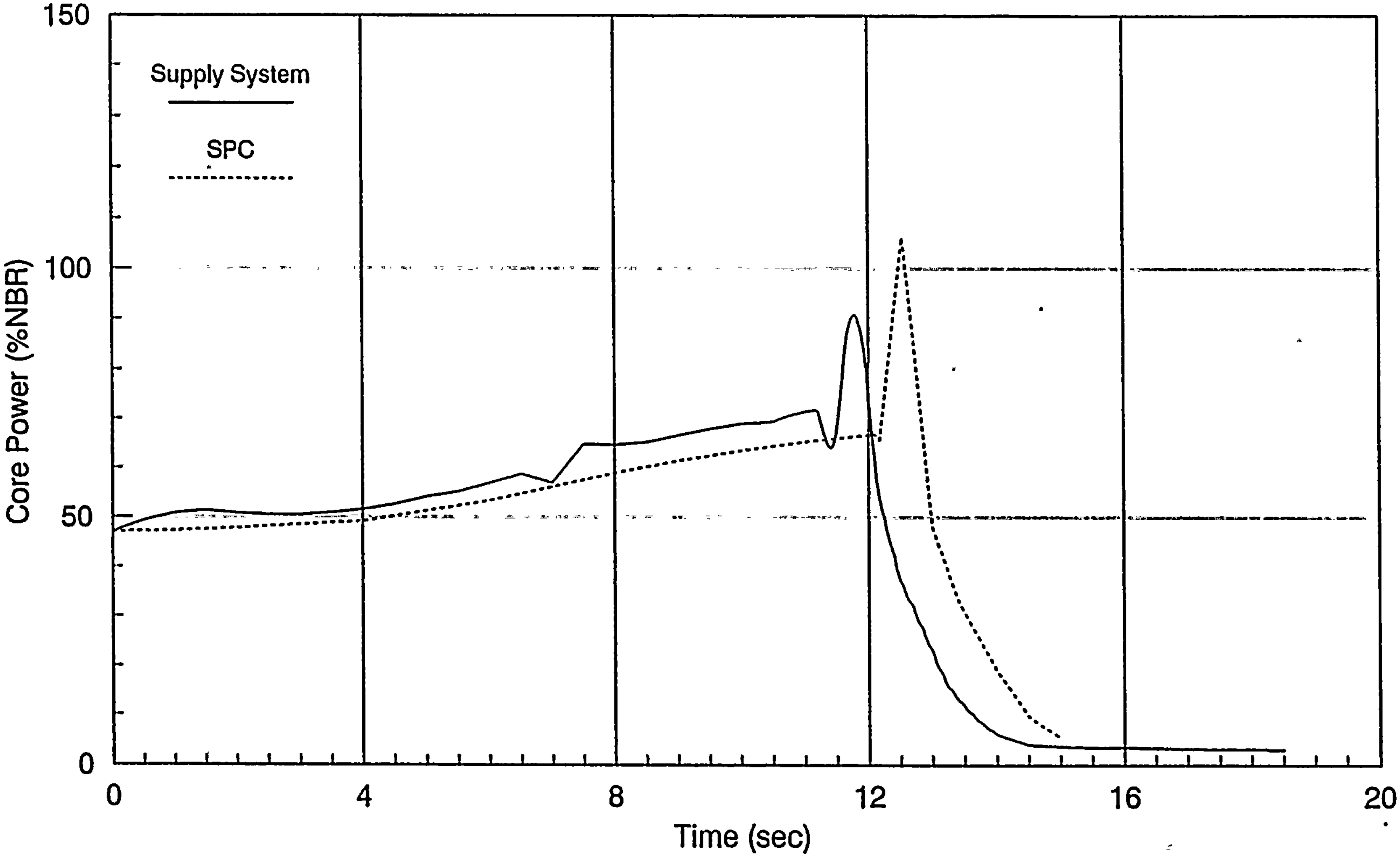


Figure 18-8 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

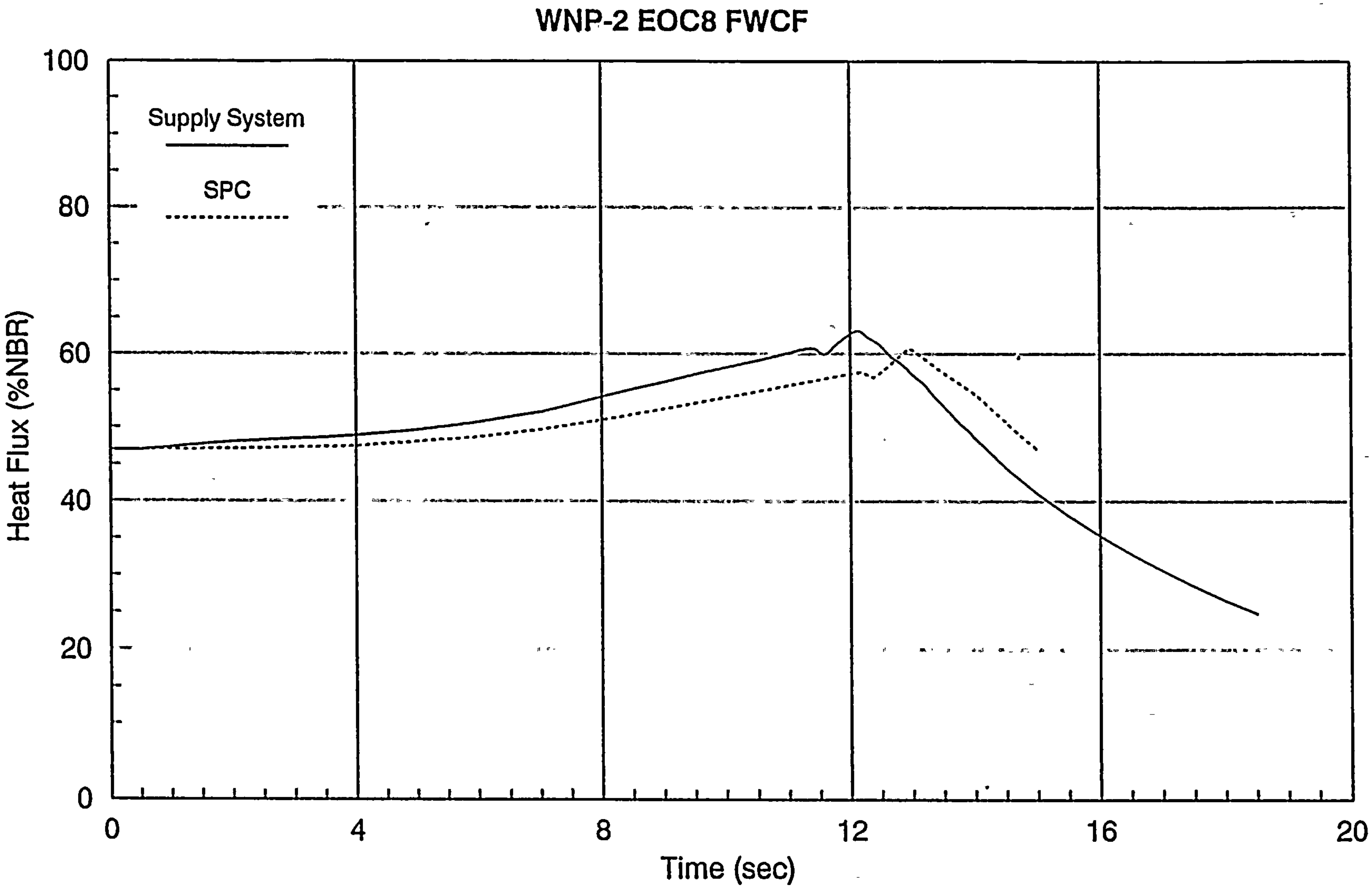


Figure 18-9 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

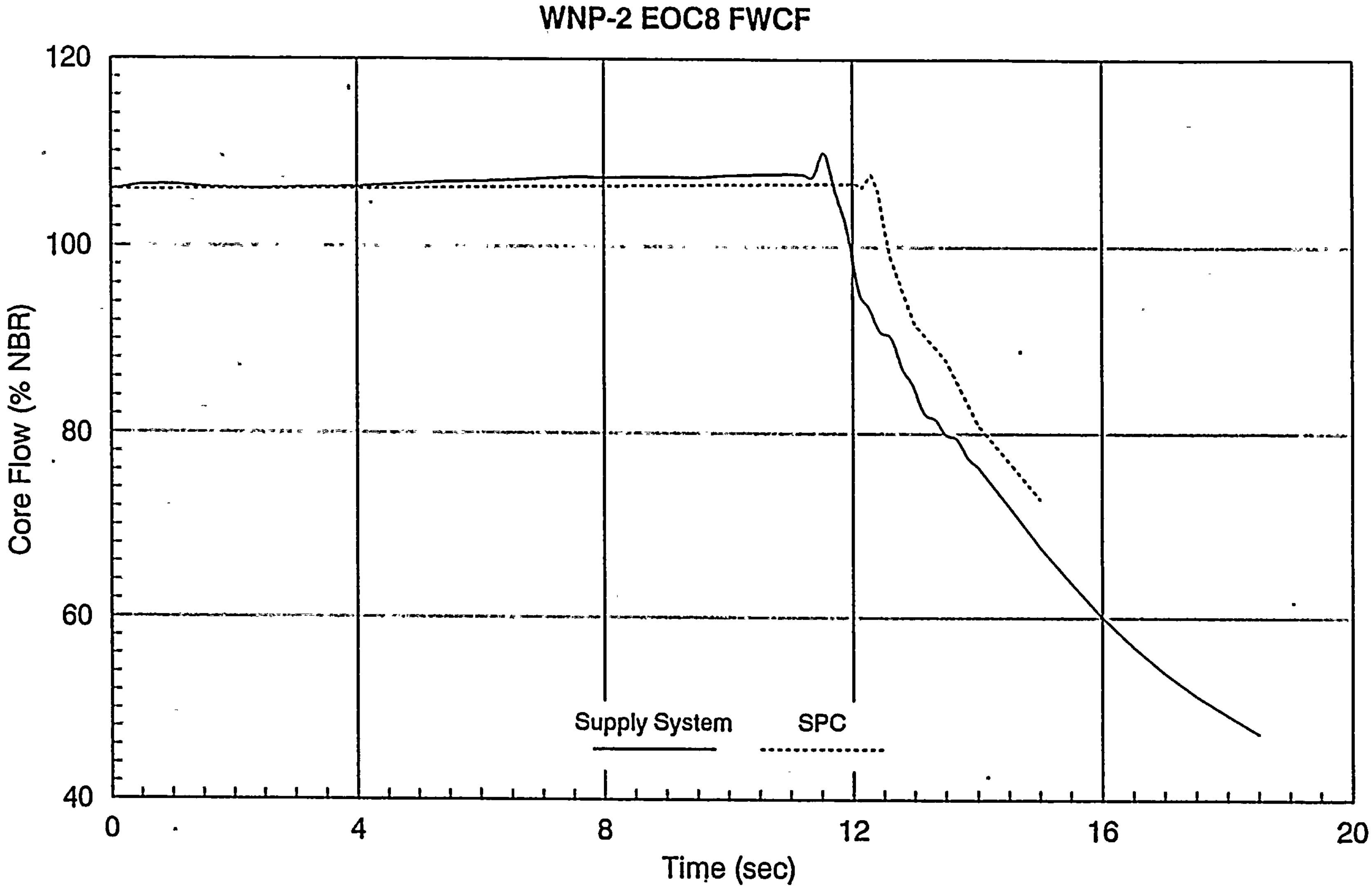


Figure 18-10 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

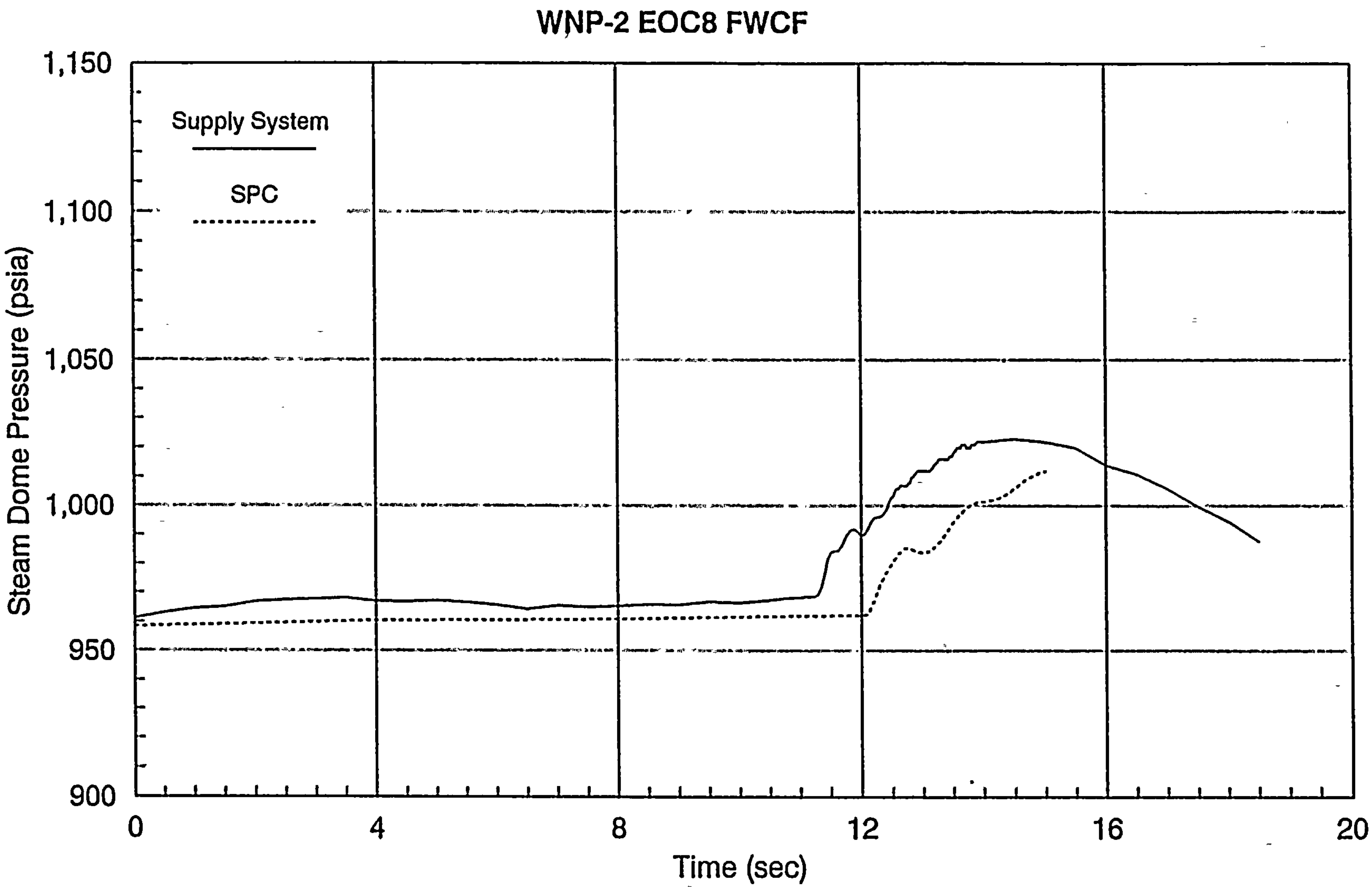


Figure 18-11 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

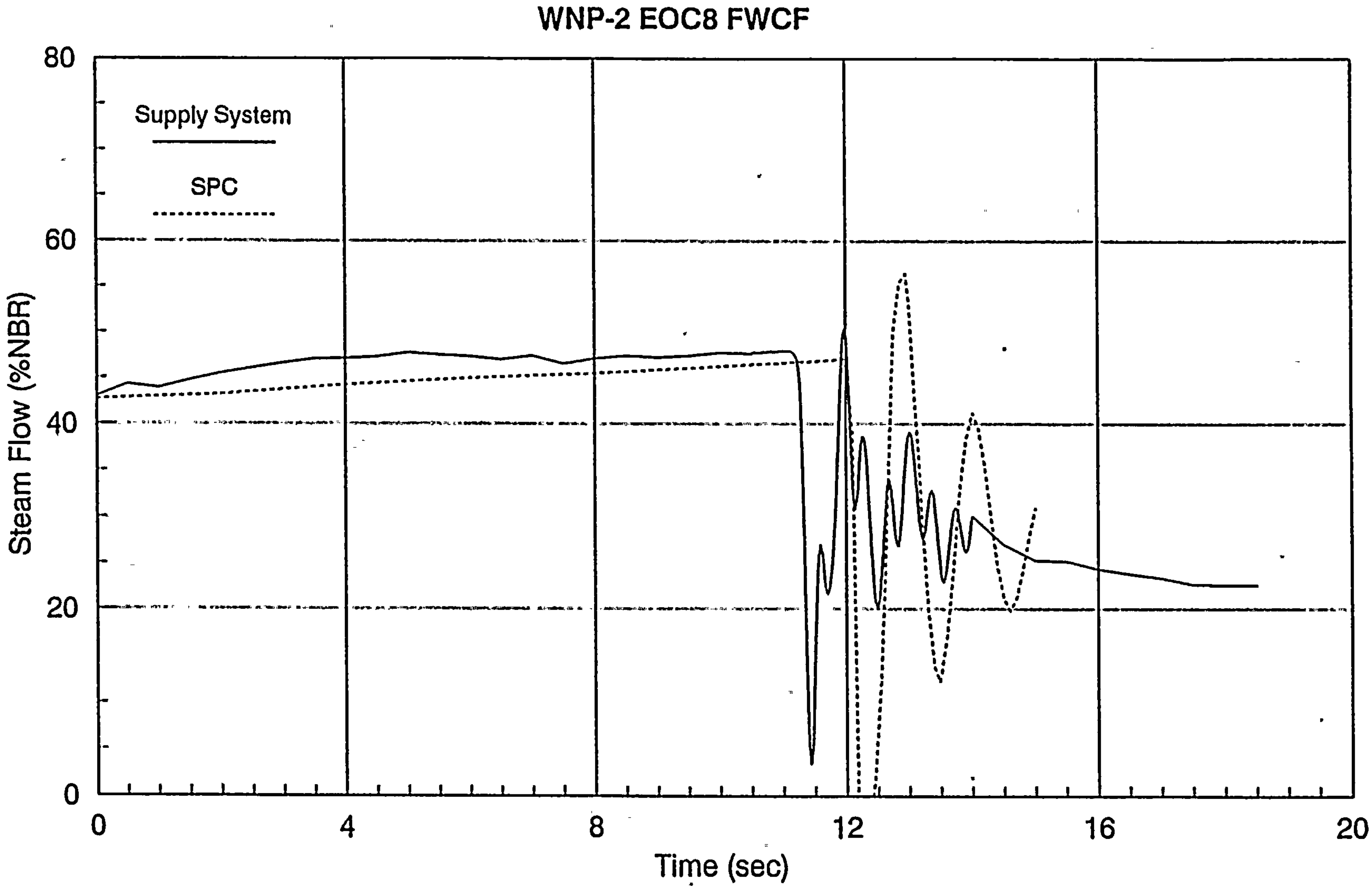


Figure 18-12 FWCF Resuts for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

WNP-2 EOC8 FWCF

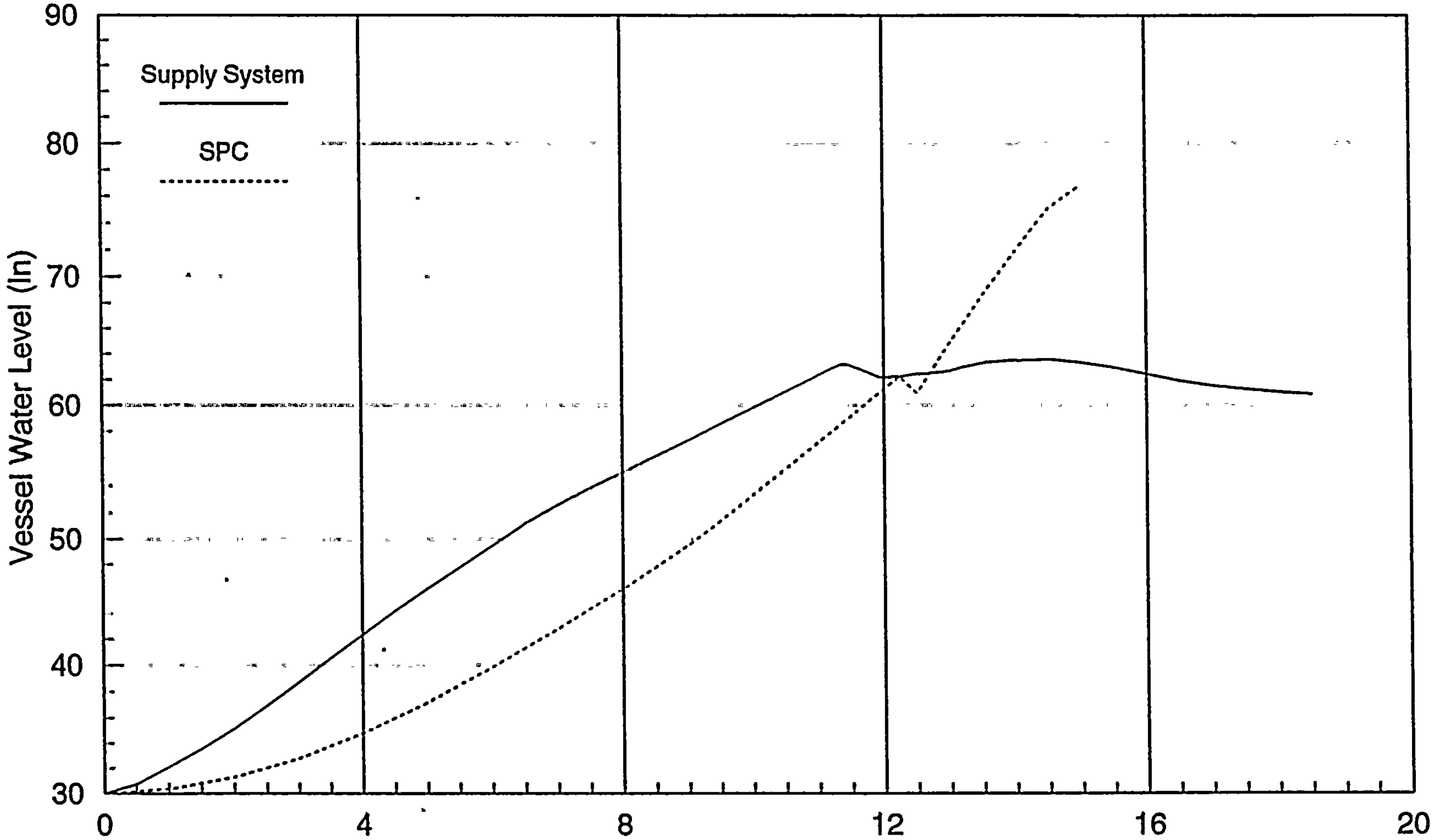


Figure 18-13 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

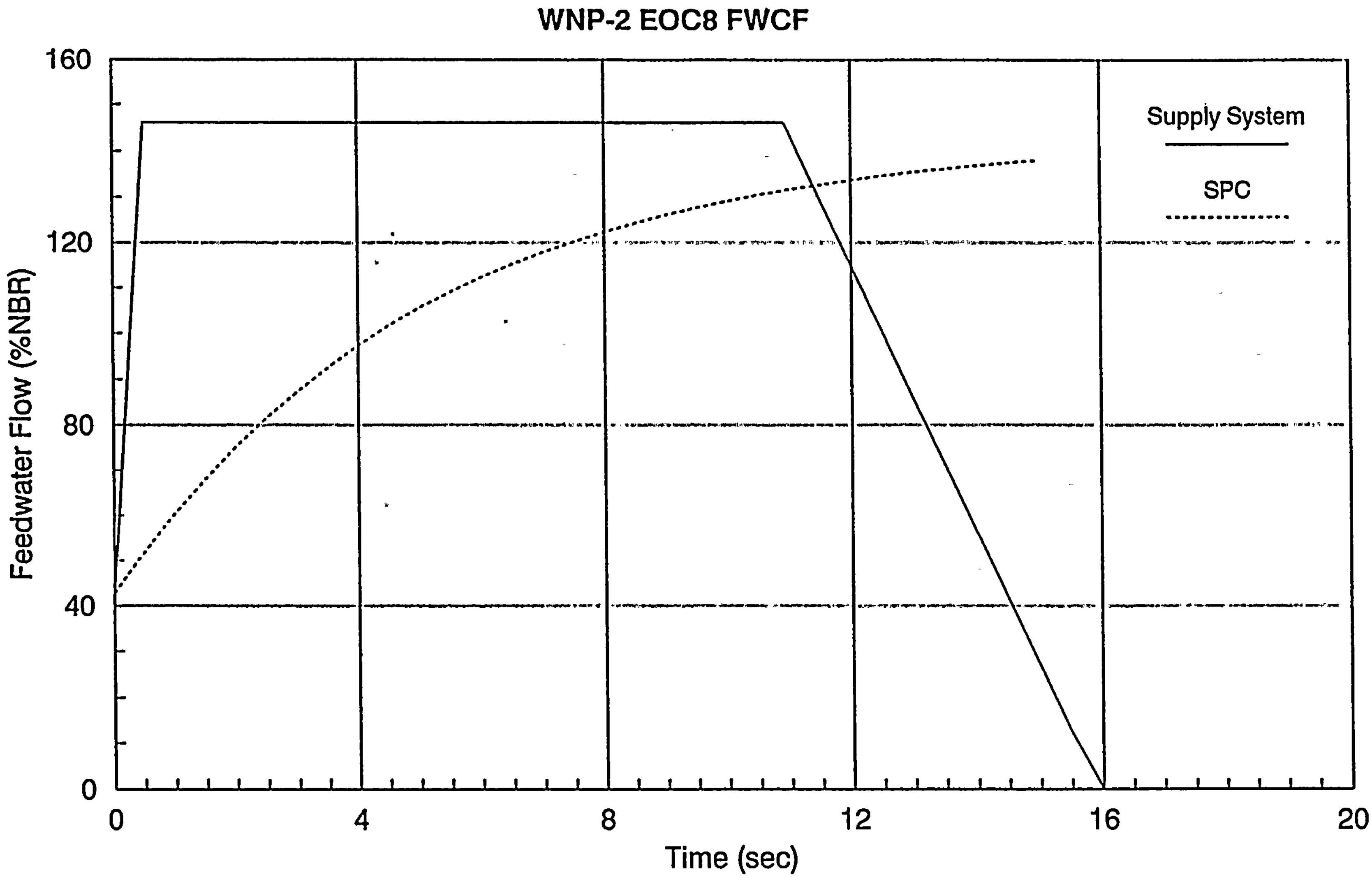


Figure 18-14 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

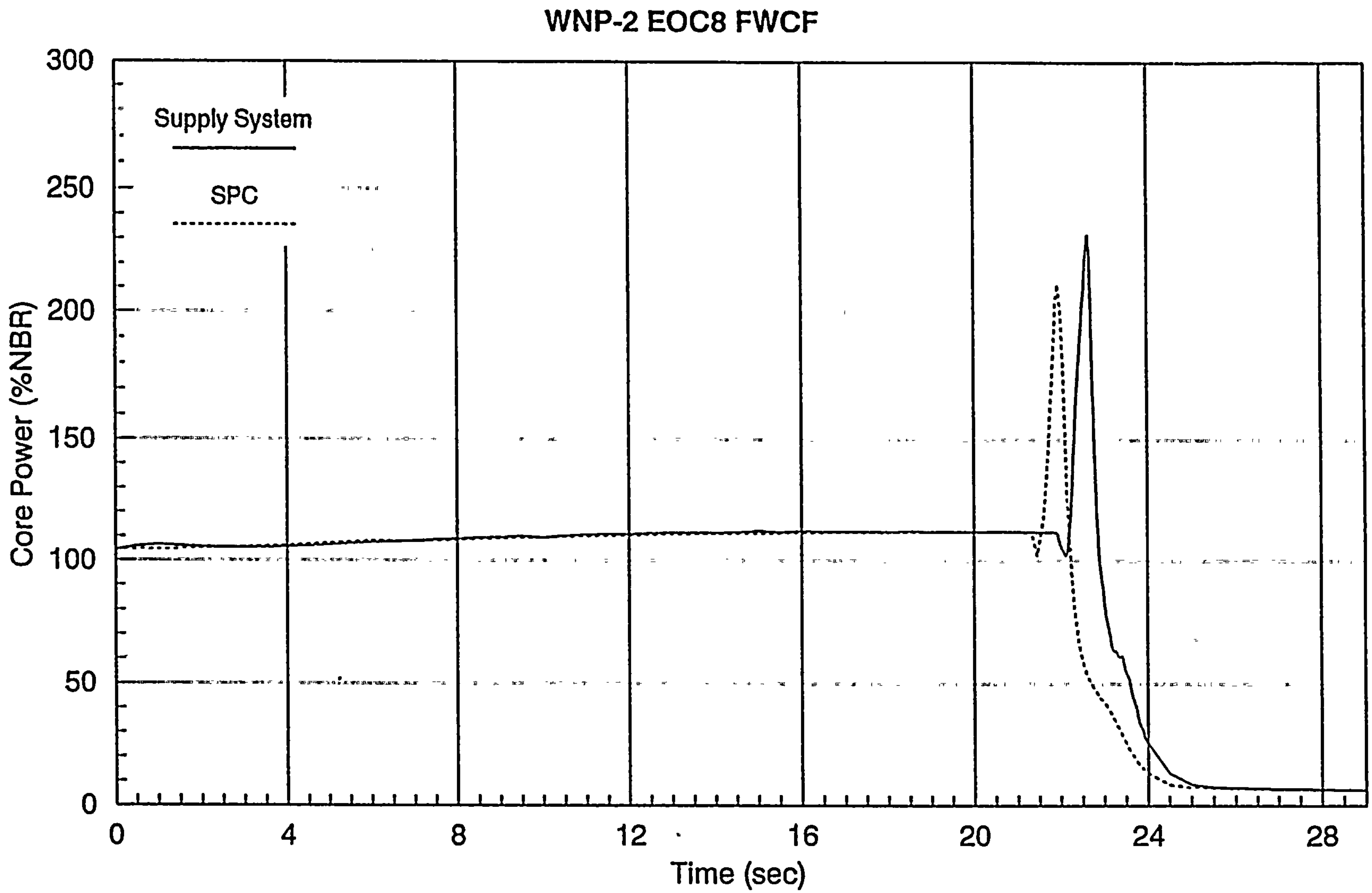


Figure 18-15 FWCF Results, 104%P/106%F
Tech. Spec. Scram Time
RPT Operable

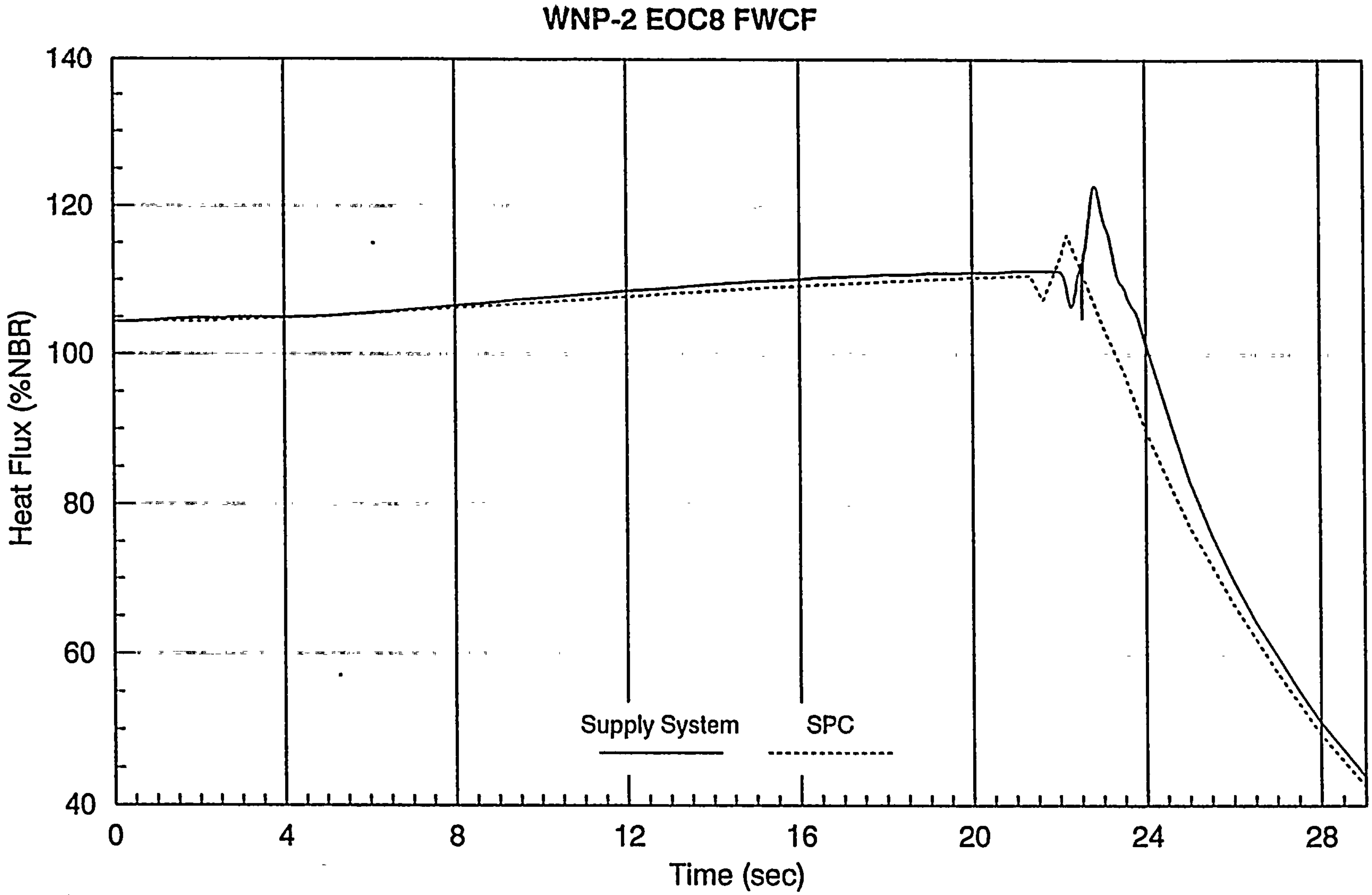


Figure 18-16 FWCF Results, 104%P/106%F
Tech. Spec. Scram Time
RPT Operable

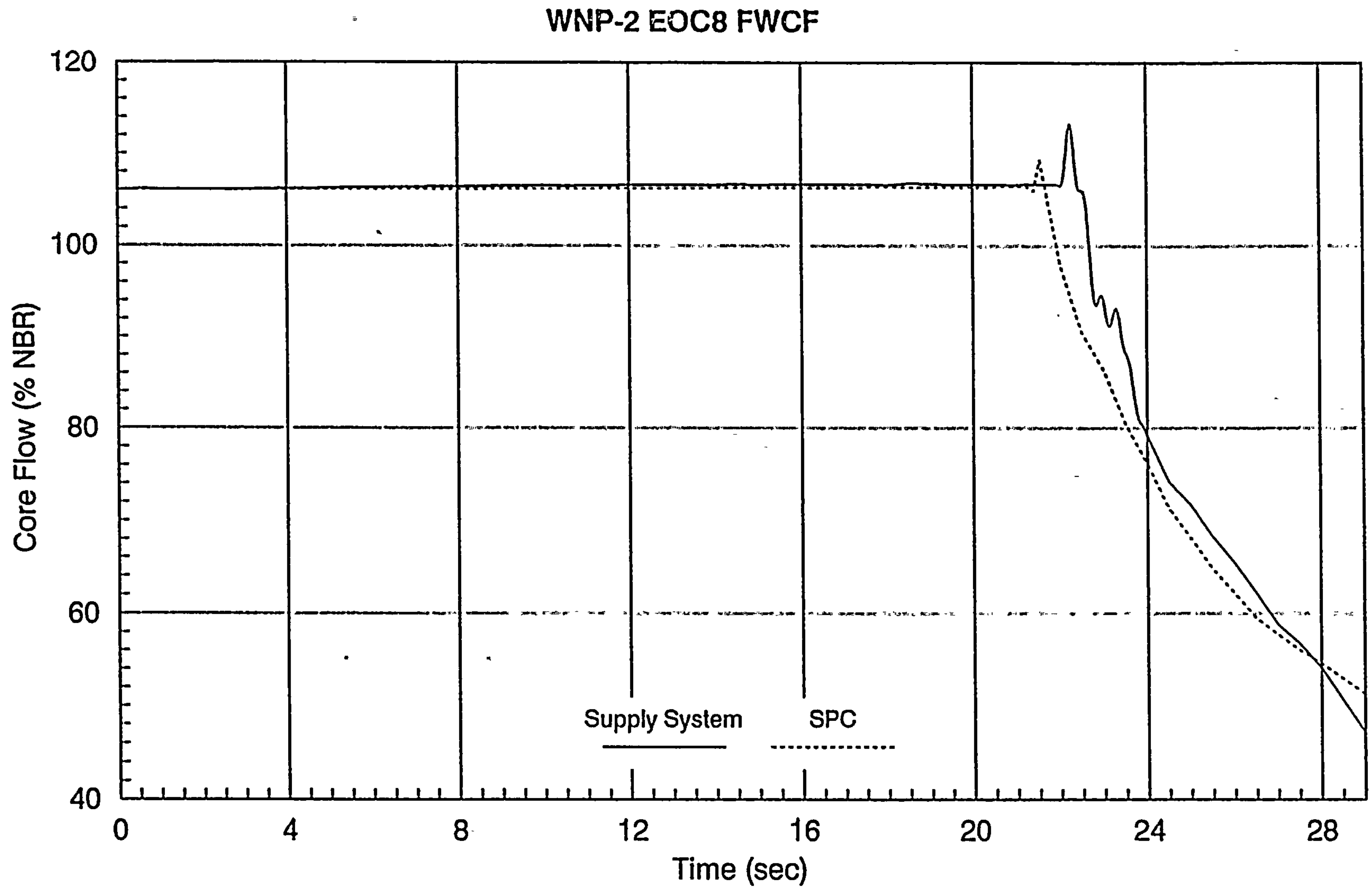


Figure 18-17 FWCF Results, 104%P/106%F
Tech. Spec. Scram Time
RPT Operable

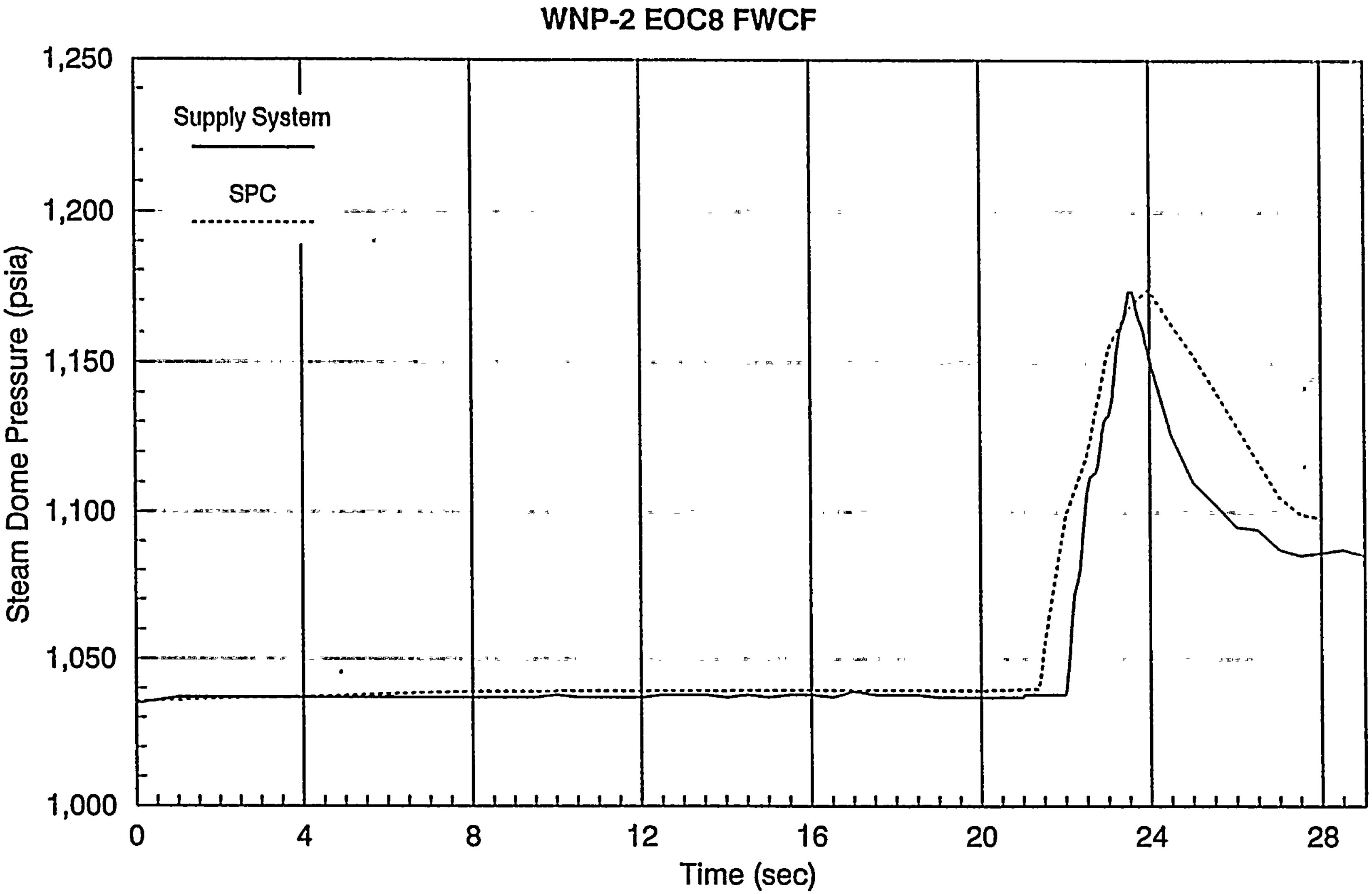


Figure 18-18 FWCF Results, 104%P/106%F
Tech. Spec. Scram Time
RPT Operable

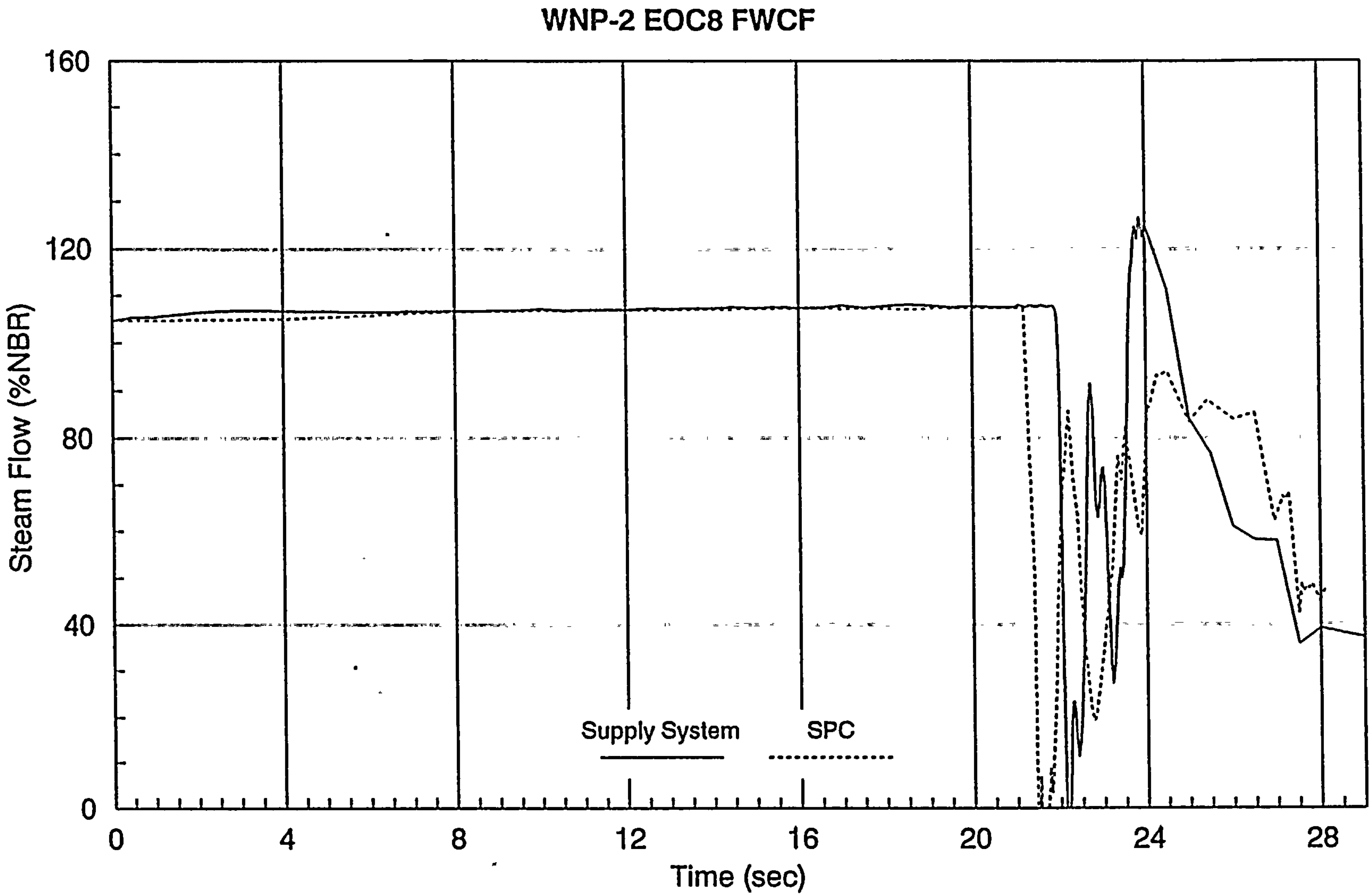


Figure 18-19 FWCF Results, 104%P/106%F
Tech. Spec. Scram Time
RPT Operable

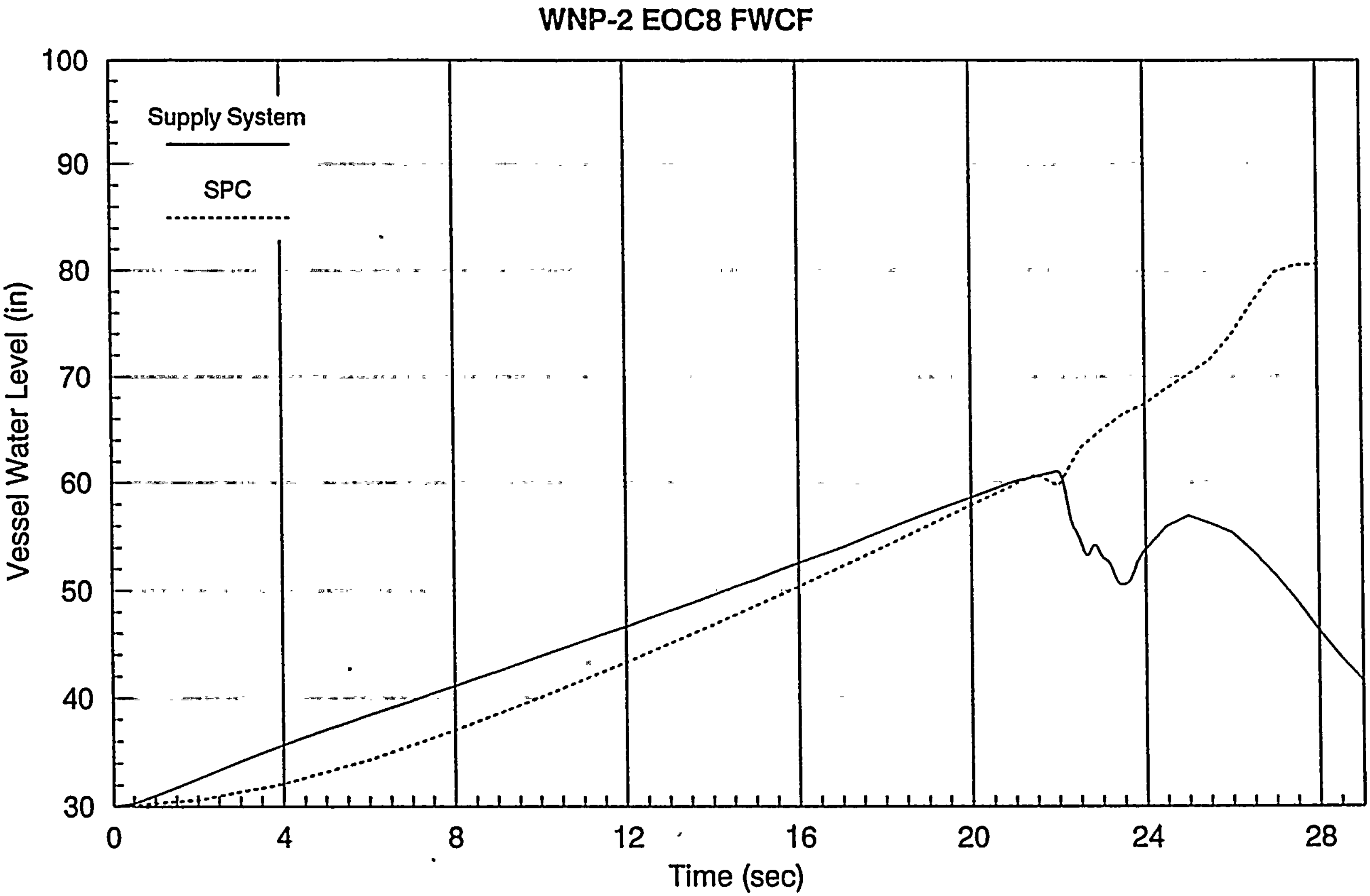


Figure 18-20 FWCF Results, 104%P/106%F
Tech. Spec. Scram Time
RPT Operable

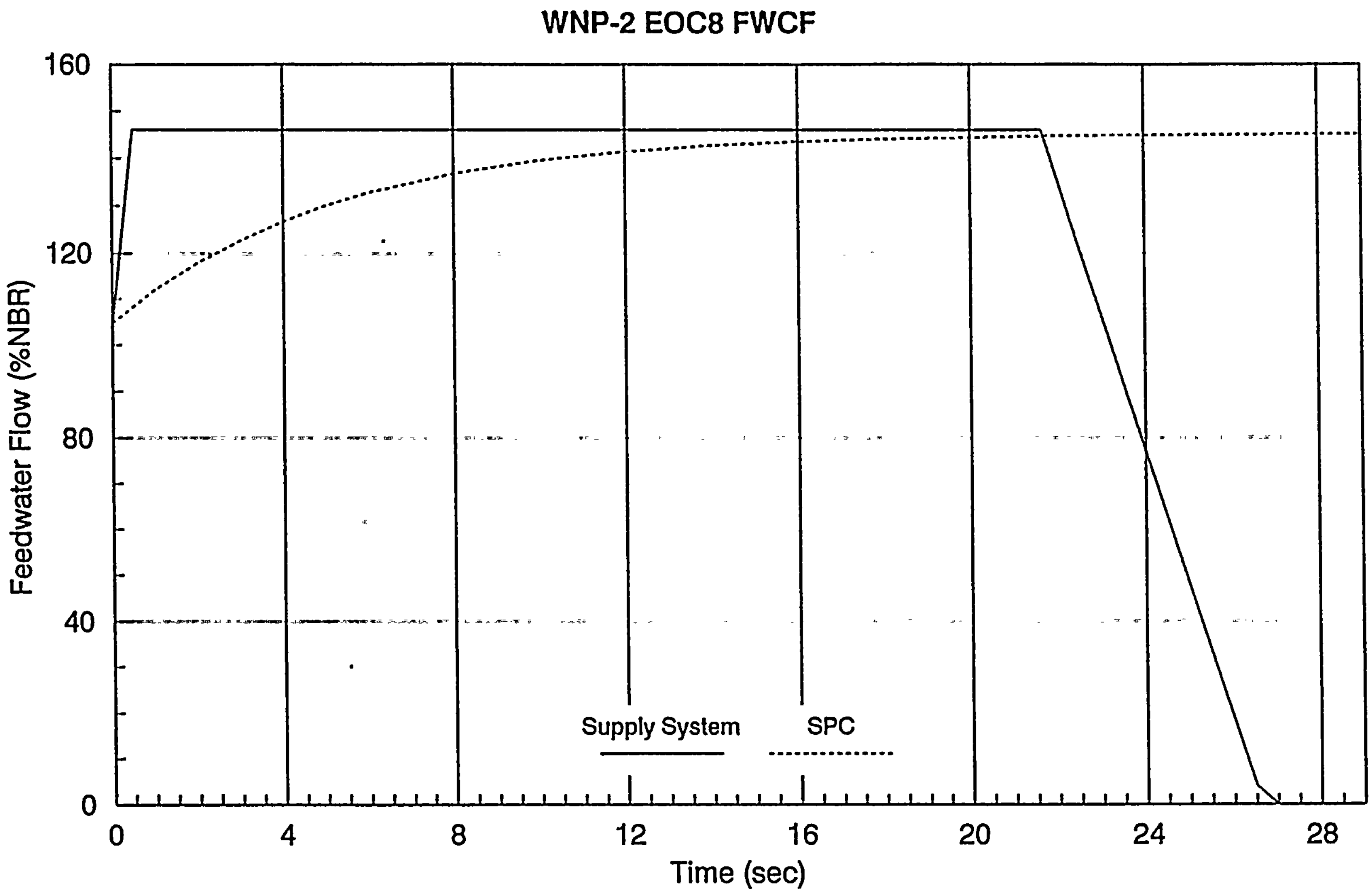


Figure 18-21 FWCF Results, 104%P/106%W
Tech. Spec. Scram Time
RPT Operable

Question 19

When performing sensitivity studies, the Supply System used a range of values for parameters considered to be important for each transient. Identify the source(s) of the ranges used in such studies.

Response

The sensitivity studies described in Appendix A of the Applications Topical Report Revision 1 are used in the determination of the overall model uncertainty. The process for determining the overall model uncertainty is described in more detail in the responses to Questions 4, 6, 7, and 10. The basis for the selection of the magnitude of the uncertainties in the individual parameters making a significant contribution to the overall model uncertainties is also provided in Appendix A of the Applications Topical Report Revision 1.

Other sensitivity studies were also performed for each transient presented in the Applications Topical. The sources of the ranges are summarized below.

1. Loss of Feedwater Heating

   As stated in the Applications Topical Report, the uncertainties in the plant operating state are included in the event acceptance limit. The fuel supplier's approved methodology (Reference 1) is used to establish this limit (i.e., Safety Limit MCPR). This limit is established by considering the uncertainties in (1) feedwater flow rate, (2) feedwater temperature, (3) core pressure, (4) core flow rate , (5) assembly flow rate, (6) radial power distribution, (7) local power distribution, and (8) CPR correlation additive constant. The sources of these uncertainties are documented in References 1 and 2. Since the SLMCPR is a function of the fuel and core design, it is updated for each cycle. It should be noted that the SLMCPR as a result of this sensitivity analysis applies to all transients when thermal limits are calculated because the parameter uncertainties and the

resulting SLMCPR were quantified independent of the transient types.

The assumption of an instantaneous 100°F feedwater temperature decrease in the loss of feedwater heating event has a considerable margin over the temperature decrease expected at the WNP-2. The temperature drop due to any single failure (e.g., bypassing feedwater heater 6) at WNP-2 is about 10-12°F (Reference 3). Therefore, the use of an instantaneous reduction of 100°F in feedwater temperature represents a significant conservatism in the event analysis. The selection of 100°F is consistent with GE (Reference 4) and the fuel vendor's (Reference 5) methodology

2. Generator Load Rejection Without Bypass (LRNB)

The uncertainty parameters studied for the LRNB transient are given in Table 5.3.3-2 and 5.3.3-3 of the Applications Topical Report, Revision 1. Uncertainty ranges for all of these parameters, except the sensitivity on RPT/no RPT, are discussed in detail in Appendix A of the Topical and the responses to Questions 4 and 10. The case for "no RPT" is to study the difference between a transient with Recirculation Pump Trip (RPT) available and a transient with RPT not available. This is not part of the model uncertainty. The case of "no RPT" is presented as a separate case in determining the Operating Limit MCPR (OLMCPR) in a reload analysis when the RPT function fails, consistent with current methodology in establishing the OLMCPRs by the fuel vendor (Reference 5).

3. Control Rod Withdrawal Error at Power

For this transient, the effect on the thermal limits from the following parameters were evaluated: (1) control rod patterns, (2) xenon distribution, (3) cycle exposure, (4) core power, (5) core flow, (6) LPRM and RBM failure modes, (7) error rod locations, (8) core pressure, and (9) core inlet subcooling. Analysis of this transient was performed by

selecting the most limiting conditions for the above parameters that would yield the worst ΔCPR. Since a conservatively bounding analysis was performed, no further sensitivity studies are required. This methodology is consistent with both fuel vendor's and GE's methodology. For detailed discussion, see Section 5.3.4 of the Applications Topical Report, Revision 1.

4. Fuel Loading Error - Mislocated Fuel Assembly

Sensitivity analysis was performed by evaluating a different combination of fuel assemblies and locations, including fresh versus once-burned fuel, high CPR versus low CPR locations. The combination with the worst ΔCPR was selected for the licensing analysis. Since a conservatively bounding analysis was performed, no further sensitivity studies are required.

5. Fuel Loading Error - Rotated Fuel Assembly

In the licensing analysis process, the important parameter is local peaking factor and is established at conservative values. Also, a rotated assembly is evaluated at each core location at each exposure step to determine the highest resulting ΔCPR over the cycle. Therefore, no further sensitivity analysis is needed.

6. Feedwater Controller Failure - Maximum Demand (FWCF)

The sensitivity cases performed in Section 5.3.7 were intended to provide additional confirmation of the FWCF model. These cases will also be performed for each cycle if continued operation of the plant is desired in the event of component failures, such as Recirculation Pump Trip failure (no RPT) discussed in Section 5.3.7.5. Therefore, they are not included in the overall model uncertainty determination.

As stated in the response to Question 4, the sensitivity analysis performed for LRNB for

establishing the overall model uncertainties is applicable to the FWCF transient.

7. Recirculation Flow Controller Failure - Increasing Flow

Similar to the Control Rod Withdrawal Error and Fuel Loading Error, the most limiting conditions are selected in the analysis of this transient. The analysis assumes that the event is initiated from the control rod line on the power flow map associated with 105% rated steam flow. The event is assumed to be terminated at 120% power and 103% core flow. The core flow at the 120% power level is based on the maximum that can be attained with the recirculation flow system design. As stated in the Applications Topical, the assumption of constant xenon bounds the conditions where xenon is allowed to build up. Since a conservatively bounding analysis was performed, no further sensitivity studies are required.

8. Control Rod Drop Accident

For each operating cycle, the Supply System evaluates the maximum rod worths, four-assembly local power peaking factors, Doppler coefficient, and delayed neutron fraction. These values are used, along with the results of the vendor's generic control rod drop accident analysis, to obtain a conservative bound for the fuel rod enthalpy deposition. Since the analysis is based on the use of NRC-approved fuel supplier methodology and the Supply System inputs to the generic parametric analysis are conservatively determined, no further sensitivity studies are required.

9. Loss of Coolant Accident

The LOCA analysis is based on the use of NRC-approved fuel supplier methodology. The required conservatism in the LOCA analysis process is based on the 10CFR50.46 event acceptance limits and the 10CFR50 Appendix K model requirements. The Supply System inputs to the LOCA analysis are determined consistent with the approved

methodology. No further sensitivity studies are required.

10. Shutdown Margin

The Supply System methodology in determining the shutdown margin has built in conservatism because it defines shutdown margin in terms of k_{ccc} (95% confidence) rather than k_{bcc} (best estimate). Because of the use of this conservative cold critical correlation, no further sensitivity studies are required.

11. Standby Liquid Control System Capability

It is assumed in the analysis that a conservatively low quantity of 660 ppm boron is present in the reactor core. Also, the analysis process assumes the most reactive point in the cycle with the smallest control rod density predicted at this condition. It is also assumed that there is no xenon present and that the most reactive reactor coolant conditions exist. This is a very conservative set of assumptions that dominates any uncertainty in the analysis process.

12. ASME Code Overpressure protection Analysis

The event definition, which assumes that the first scram initiation signal encountered is inoperable, provides a substantial amount of conservatism in the analysis process. Further, the ASME Code incorporates additional margin in establishing the limits used in the analysis process. No further sensitivity studies are required.

13. Stability

NRC-approved analysis methods are used to evaluate decay ratios at selected operational points on the power/flow map. Recommendations of the BWR Owners Group Committees on Stability and Reactivity Control, and the NRC guidelines (e.g., Generic

Letter 94-02, Reference 6), will be followed in regard to the required analysis, including sensitivity analysis.

REFERENCES

1. Advanced Nuclear Fuel, SAFLIM2, A Monte Carlo Simulation Code for Analysis of Fuel Rods in Boiling Transition in a BWR Core, Users Manual, ANF-1261(P), Revision 0, August 1989

2. B.M. Moore, Cycle 8 Safety Limit, Calculation File NE-02-91-40, Washington Public Power Supply System, March 1992

3. Private communication, P. Bentrup (Supply System), June 1994.

4. Washington Public Power Supply System, WNP-2 Final Safety Analysis Report, Amendment 48, August 1993

5. J.G. Ingham, WNP-2 Cycle 8 Plant Transient Analysis, EMF-92-039, Rev. 1, Siemens Nuclear Corp., June 1992

6. USNRC Generic Letter 94-02, Long Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors, July 11, 1994

Question 20

Identify any transients for which system actuation on low reactor water level is relied upon. If there are any, justify the ability of the RETRAN model to compute a conservative water level for this purpose.

Response

In the Supply System plant safety analysis, credit for system actuation on low reactor water level may be taken for any anticipated operational occurrence (transient) that results in the loss of feedwater. The system actuations on low water level for anticipated operational occurrences may include reactor scram, initiation of the high pressure water makeup systems, and reactor vessel isolation. As demonstrated by the WNP-2 safety analysis documented in Section 15 of the Final Safety Analysis Report, the events that experience a low reactor water level are relatively mild from the perspective of the challenge to event limits or acceptance criteria. None of the potentially limiting anticipated operational occurrences (events that have the potential to establish core operating limits) analyzed using RETRAN-02 take credit for system actuations on low reactor water level. Therefore, the core operating limits established using RETRAN-02 are not sensitive to the prediction of low reactor water level.

As discussed in the response to Question 18, Feedwater Controller Failure (FWCF) is the only potentially limiting transient that takes credit for system actuation on high water level. Underprediction of water level makes the thermal limit calculation for FWCF conservative. As discussed in the response to Question 18, the difference in water level translates into a 1% increase in core peak heat flux, which is equivalent to an increase of about 0.01 in ΔCPR.

Water level calculation in the WNP-2 RETRAN model is based on the total collapsed water volume in the upper, middle and lower downcomers (Volumes 20, 18 and 19 respectively, see Reference 1) as calculated by RETRAN, but is modified to account for the cross sectional area variations in these downcomer regions. In the WNP-2 RETRAN model for the collapsed water

volume, the cross sectional areas for each of the three downcomer regions are considered to be uniform. This is an intrinsic feature of the RETRAN model. The water level as calculated using the uniform cross sectional areas does not reflect the variable cross sections. Consequently, a control system was built in the WNP-2 RETRAN model to allow a table lookup to account for cross sectional area variation and thus obtain the corrected water level. This table consists of a set of elevation versus volume values which allows for cross sectional area changes in the standpipes, steam separators, separator annuli and space between separators and steam dryer. The output of the control system gives a water level that is further adjusted. An adjustment is applied to the water level to correct for the pressure drop in the steam dryer because the steam dryer is combined with the steam dome as one volume in the RETRAN model. At rated condition, it is an 8.8-inch correction. This corrected water level is used for the trip functions in other parts of the RETRAN model.

REFERENCE

1. Y.Y. Yung et al., BWR Transient Analysis Model, WPPSS-FTS-129, Rev. 01, Washington Public Power Supply System, September 1990.

Question 21

The Supply System, where applicable, compared its results against those of ANF. Please identify the version of COTRANSA used, and if it is COTRANSA and not COTRANSA1, identify the bias in COTRANSA vs. the plant data.

Response

The version of COTRANSA used in comparison with Supply System calculations is COTRANSA2 Version UAPR91.

Question 22

Although the Supply System stated that the ANF approved LOCA methodology is incorporated into the Supply System LOCA methodology, it does not explain how the two methodologies are integrated and used. Provide a detailed discussion of how this is accomplished

Response

The loss of coolant accident (LOCA) analysis is performed in the plant safety analysis process to demonstrate compliance to the requirements of 10CFR50.46. In the LOCA analysis process, input assumptions are made regarding the nuclear kinetics parameters and the reactor operating state prior to the event. Conformance with these analysis input assumptions is demonstrated for each reload core design to provide assurance that an unanalyzed plant operating condition does not exist.

The methodology and licensing analysis process for demonstrating the compliance to the event acceptance limits for the loss of coolant accident (LOCA) are described in Sections 5.3.10.3 and 5.3.10.4 of the Applications Topical Report Revision 1. This process has been developed to demonstrate that the bounding assumptions made in the LOCA analysis to establish the core operating limits for the individual fuel designs by the reload fuel supplier are applicable to the new reload core designs.

The neutronics parameters are the key LOCA analysis parameters that can change as a result of the reload core design, assuming that a LOCA heat up analysis has been performed for the specific fuel types utilized in the reload core design. It should be noted that a LOCA heatup analysis is required by the WNP-2 reactor analysis methodology to be performed, using NRC approved LOCA analysis methodology, to establish the core operating limits for each fuel type in the core. These neutronics parameters include: (1) the local peaking factors; (2) fuel rod power history; (3) void reactivity, (4) Doppler reactivity; and (5) scram reactivity. The local

peaking factors and fuel rod power history are provided as direct inputs in the LOCA analysis process to establish the operating limits for specific fuel designs. The remaining three reload cycle specific parameters are compared to the bounding analysis input assumptions to demonstrate conformance to the analysis input assumptions.

The codes used to calculate these five neutronic parameters include: (1) MICBURN-E; (2) CASMO-2E; (3) NORGE-B; (4) SIMULATE-E; (5) SIMTRAN-E; and (6) RETRAN-02. The specific codes used to generate the cycle specific values for each of these neutronics parameters are provided in Section 5.3.10.3 of the Applications Topical Report Revision 1. The specific steps for determining the appropriate cycle specific values are provided in Section 5.3.10.4 of this report. The calculated results for Cycle 8 are described in Section 5.3.10.5 of this report.

Question 23

Identify any transients for which reverse jet pump flow is predicted. For those transients, qualify the jet pump model in that regime.

Response

As demonstrated by the WNP-2 safety analysis documented in Section 15 of the Final Safety Analysis Report, the events that experience reverse jet pump flow are relatively mild from the perspective of the challenge to event limits or acceptance criteria. None of the anticipated operational occurrences that have the potential to establish core operating limits and that are analyzed using RETRAN-02 experience reverse jet pump flow. Therefore, the core operating limits established using RETRAN-02 are not sensitive to the prediction of reverse jet pump flow.

Reverse jet pump flow is possible in transients for which the pump in one of the recirculation loops is not operating (single loop operation). The one pump trip Power Ascension Test (PAT 30A) documented in the BWR Transient Analysis Model Topical Report WPPSS FTS-129, Rev. 1 is an example of such a transient. In this event, the flow through the jet pumps in the loop with the tripped pump reverses from positive to negative flow during the course of the transient.

The responses to Question 1.5 and Question 2.3 of Reference 1 and Question 3 of Reference 2 involve the RETRAN-02 analysis of PAT 30A, and demonstrate that RETRAN-02 can correctly compute reverse flow in a jet pump. In addition, the code prediction of the resulting core power level and jet pump flow rates were in good agreement with the measured test data.

The SER for the BWR Transient Analysis Model Topical Report WPPSS FTS-129, Rev. 1 recommended that the reverse jet pump flow condition be considered for the transient MCPR. In situations where one recirculation pump is operating while the other is tripped, the reactor reaches a steady-state power level that is approximately 73% (or less) of full power with a

reduced core flow. This power level and the corresponding flow condition are not limiting operating conditions with respect to either the MCPR or other plant operating conditions.

In summary, the licensing basis model for WNP-2 using RETRAN-02 is conservative for the application where a reverse flow is anticipated for the following reasons:

(1) No reverse jet pump flow is anticipated for any of the three potentially limiting transients (Load Rejection Without Bypass, Turbine Trip Without Bypass, and Feedwater Controller Failure) under the normal two-loop operation mode.

(2) For the case of single loop operation, where a slight reverse flow in the idle loop exits, the power level will be limited to 73% rated or less with a reduced core flow. Generic analysis by the Supply System and the fuel supplier indicate that the operating limits determined by two-loop operation bound the single loop operation.

(3) Power Ascension Test 30A, where one recirculation pump is tripped off, supports the RETRAN model for WNP-2 in correctly predicting the reversed flow through the jet pumps in the idled loop.

REFERENCES

1. Letter, G02-91-134, G.C. Sorensen to NRC, <u>Response to Request for Additional Information Regarding Topical Report WPPSS FTS-129, BWR Transient Analysis Model (TAC No. 77048)</u>, July 15, 1991.

2. Letter, G.C. Sorensen to NRC, <u>Response to Second Request for Additional Information Regarding Topical Report WPPSS FTS-129, BWR Transient Analysis Model (TAC No. 77048)</u>, March, 1992.

Question 24

Describe the "licensing" model for the loss of FW heating event including the power distribution used.

Response

The "licensing" model for the loss of FW heating event was described in Section 5.3.2 of the Applications Topical Report Revision 1. The power distribution was calculated by SIMULATE-E. The use of SIMULATE-E for modeling this event is consistent with the current vendor's, Siemens Power Corporation, appoach. Further description of the loss of FW heating event is provided in Question 25.

Question 25

Some reload analyses were performed as steady state events using SIMULATE-E, a reactor physics code. The stated justification is based on the fact that these events take place "very slowly" therefore the core remains in a quasi-steady-state condition. Provide and justify determination criteria for slowness of events to be analyzed as steady state. Furthermore, qualify SIMULATE-E for this type of applications and justify not using a T/H code.

Response

The Supply System methodology includes quasi-static analyses for the Loss of Feedwater Heating (LFWH) and Control Rod Withdrawal Error (CRWE) events. In each of these events, the use of a three-dimensional, steady state analysis results in a greater level of detail and therefore a more mechanistic analysis than either a point kinetics or a one-dimensional transient analysis. Such methods are in use at domestic fuel vendors (References 1, 2 and 3) and SIMULATE-E has been accepted for licensee analyses of this type (References 4, 5 and 6). As is discussed in References 2 and 3, Siemens Power Corporation, the current vendor, uses this method.

Criteria for Determination of Slow Events. In current generation transient analyses, two-dimensional, radial effects are assumed to remain constant during the course of the event. This assumption is effected by the use of a constant ratio of hot channel power to core power. While such an assumption is valid for events which progress rapidly enough that radial power redistribution effects are negligible (such as the pressurization events, which generally terminate within two or three seconds), events which proceed slowly enough that power feedback effects on coolant flow are realized are not simulated accurately with this assumption. In general, power feedback on coolant flow is fully realized at about six seconds after the power perturbation. Although three-dimensional power redistribution within the core occurs for all anticipated operational occurrences, transient events which have less than 25% core power remaining after three seconds of event time have effectively terminated prior to any appreciable

radial power redistribution and may be modeled adequately using point or one-dimensional kinetics techniques.

Justification for Using Quasi-Static Methods. Potentially limiting events which result in early scram, such as the Load Rejection or Turbine Trip Without Bypass transients (LRNB/TTNB) and the Feedwater Controller Failure to Maximum Demand transient (FWCF), are characterized by early scram, rapid upward and downward changes in power level, and low power at late event times. Calculated ΔCPR is strongly dependent on the relative timing of the power transient and the flow transient, which are proceeding in directions of opposing severity at the time of minimum CPR. Missing either power or flow timing only slightly may result in a large change in calculated ΔCPR, so event consequences are appropriately calculated with an analytical tool designed to model the temporal aspects accurately.

The events which require three-dimensional analysis are characterized by slower changes in core power level, high power levels late in the transient, and effective core state equilibration at the point of minimum CPR. In these events, missing the timing of a power or flow change has little impact on the calculated final result. The determining factor is the redistribution of power in the core, which is available only in three dimensional analysis.

Loss of Feedwater Heating. In the LFWH event, core power ascends slowly until a new steady state operating condition is reached. Use of quasi-static methods allows the end point to be assumed without calculating the otherwise unimportant rate of power ascension.

In transient analysis of the LFWH event, the hot channel analysis determines the ΔCPR of a single bundle during the transient. In the physical event, however, three-dimensional power redistribution effects may result in different bundles experiencing the least CPR margin at the beginning and end of the transient. The quasi-static analysis evaluates these power redistribution effects so that the ΔCPR can be appropriately defined on the basis of all the bundles of a given type in the core.

Use of quasi-static methods conservatively neglects any scram set point that maybe encountered, and the steady state analysis provides an accurate evaluation of differences in Doppler and void response between different bundles in the core.

Control Rod Withdrawal Error. The rationale behind the use of three-dimensional, quasi-static analysis for the CRWE evaluation is similar to that of the LFWH analysis. In the physical CRWE event, almost all of the effects are localized because only one control rod is moved. Use of point or one-dimensional simulation of this type of event does not allow calculation of such regional response, while the three-dimensional analysis allows an accurate prediction of localized power, hydraulic, and instrument response effects.

In the transient analysis of the CRWE event, the movement of a single control rod is modeled by a very small change in control rod density at each axial node. Corewide effects of this change are very small; translation of these effects to the hot channel analysis results in negligible event consequences. In the quasi-steady state analysis, withdrawal of a single control rod results in a substantial redistribution of core power toward the region of the withdrawn rod, resulting in increased power, perturbed individual channel flow, and changed local instrument response. This analysis methodology provides a much more accurate simulation of the event.

The quasi-static CRWE analysis assumes prompt production of all neutrons, which results in a higher localized power than would be experienced in the physical event, in which a fraction of the neutrons would be delayed up to several seconds. The quasi-static analysis also assumes immediate equilibration of flow effects, which results in a lower flowrate in the perturbed assemblies than would be experienced in the physical event, in which flow equilibration would take several seconds. This combination of higher power and lower flow results in calculation of incremental ΔCPR values which are higher than those in the physical event. Performance of the Rod Block Monitor (RBM) may be addressed within the quasi-static analysis. This follow-on analysis allows the added evaluation of instrument failures within the RBM system, which has an impact on the calculated ΔCPR. This option is not available with one-dimensional simulation of the event.

<u>Qualification of SIMULATE-E.</u> The accuracy with which SIMULATE-E predicts three-dimensional power distribution effects is assessed in the Supply System's approved physics methods topical report (Reference 7). The Supply System implementation of SIMULATE-E predicts power distribution effects with a power distribution uncertainty factor consistent with the fuel vendor and other licensees.

The power distribution benchmark includes prediction of instantaneous effects (incore instrument responses) and irradiation effects (gamma scan measurements). These benchmarks demonstrate the ability of the program to predict power distributions accurately within the three-dimensional core simulation.

REFERENCES

1. General Electric Standard Reload Application for BWR Fuel, NEDO-24011-A, General Electric Company.

2. T.L. Krysinski et al., Exxon Nuclear Methods for Boiling Water Reactors: Neutronics Methods for Design and Analysis, XN-NF-80-19(A), Volume 1, Exxon Nuclear Company, Inc. 1980; proprietary.

3. R.G. Grummer and J.C. Rawlings, The Loss of Feedwater Heating Transient in Boiling Water Reactors, ANF-1358(A), Siemens Power Corporation, October 1990.

4. A. Dyszel, K.C. Knoll, J.H. Emmett, E.R. Jebsen, C.R. Lehmann, A.J. Roscioli, R.M. Rose, J.P Spadaro, and W.J. Weadon, Qualification of Steady State Core Physics Methods for BWR Design and Analysis, PL-NF-87-001-A, Pennsylvania Power & Light Company, 1987.

5. S.R. Hesse, Methods for Performing BWR Steady-State Reactor Physics Analysis, PECO-FMS-0005-A, Volume 1, Philadelphia Electric Company, 1988.

6. C.H. Greene, D.C. Albright, B.J. Boyle, J. Juneau, L.A. Leatherwood, G.W. Scronce, P.V. Vo, and M.L. Wittenburg, Steady State Core Physics Methods for BWR Design and Analysis, EA-CA-91-001-M-A, Gulf States Utilities Company, April 1993.

7. B.M. Moore, A.G. Gibbs, J.D. Imel, J.D. Teachman, D.H. Thomsen, and W.C. Wolkenhauer, Qualification of Core Physics Methods for BWR Design and Analysis, WPPSS-FTS-127(-A), Washington Public power Supply System, March 1990.

Question 26

For the Generator Load Rejection transient, the Supply System reduced a number of independent variables in the response surface equation to one, the control rod scram time. This may be an over-simplification of the response surface method. Justify not considering other variables in the equation.

Response

The Statistical Combination of Uncertainties (SCU) methodology, which incorporates the use of a response surface, is shown schematically in Figure 26-1. In general, the uncertainties associated with one or more input parameters are treated statistically while other uncertainties are accounted for by the overall model uncertainty. The statistical simulation of an event requires that a large number of cases be run in order to obtain the desired accuracy. Use of a response surface is an efficient way of reducing the number of required system and core thermal-hydraulic analyses required to make accurate statistical statements about the event analysis results. As shown in the answer to Question 5, the response surface used by the Supply System is of the general form

$$Y_{rs} = B_0 + \sum_{i=1}^{N} B_i \, \eta_i + \sum_{i=1}^{N} B_{ii} \left(\eta_i^2 - C \right) + \sum_{i=1}^{N} \sum_{j=1,\, j<i}^{N} B_{ij} \eta_i \, \eta_j$$

where:

B_o, B_i, B_{ii}, and B_{ij} are response surface fitting coefficients;
η_i are the response surface independent variables;
N is the number of independent variables;
C is a known normalization coefficient that provides orthogonal polynomials;
Y_{rs} is the response surface dependent variable

The function of a response surface is to provide an approximation of the event analysis results that would be obtained from calculations using RETRAN-02 and VIPRE-01. The number of input parameters that are treated statistically determines the form of the response surface and hence the number of coefficients required to specify the response surface. A single input parameter treated statistically would produce a two-dimensional response surface, two parameters would produce a three-dimensional surface, and three or more parameters would produce a higher dimension hyper surface. The validity of a particular response surface is determined by how well it fits the calculated results. There is no intrinsic requirement for any particular number of independent parameters.

The Supply System chose to treat only the control rod scram time as an independent parameter in the response surface. This choice was based on the sensitivity of calculated RCPR to scram time and the availability of a large scram time data base that allows accurate quantification of the scram speed statistics. It should be noted that the normalization coefficient, C, was set to 0 for this response surface since orthogonal polynomials are not relevant to a single parameter response surface and including coefficient C would not change the fit accuracy. Any contribution of coefficient C is included in the constant term B_o. Therefore, for the Supply System application, the response surface is the polynomial:

$$RCPR = B_o + B_I T + B_{11} T^2$$

As shown on Figure 26-2, this single parameter response surface provides an excellent fit to the calculated RCPR results over the entire range of scram times that were simulated in the RETRAN-02 analyses. Other input parameters whose uncertainties could have been chosen as independent variables are conservatively accounted for in the overall model uncertainty as shown in Tables A-1 and A-2 of the Applications Topical Report Revision 1.

Figure 26-1
Supply System Statistical Combination of Uncertainties Methodology

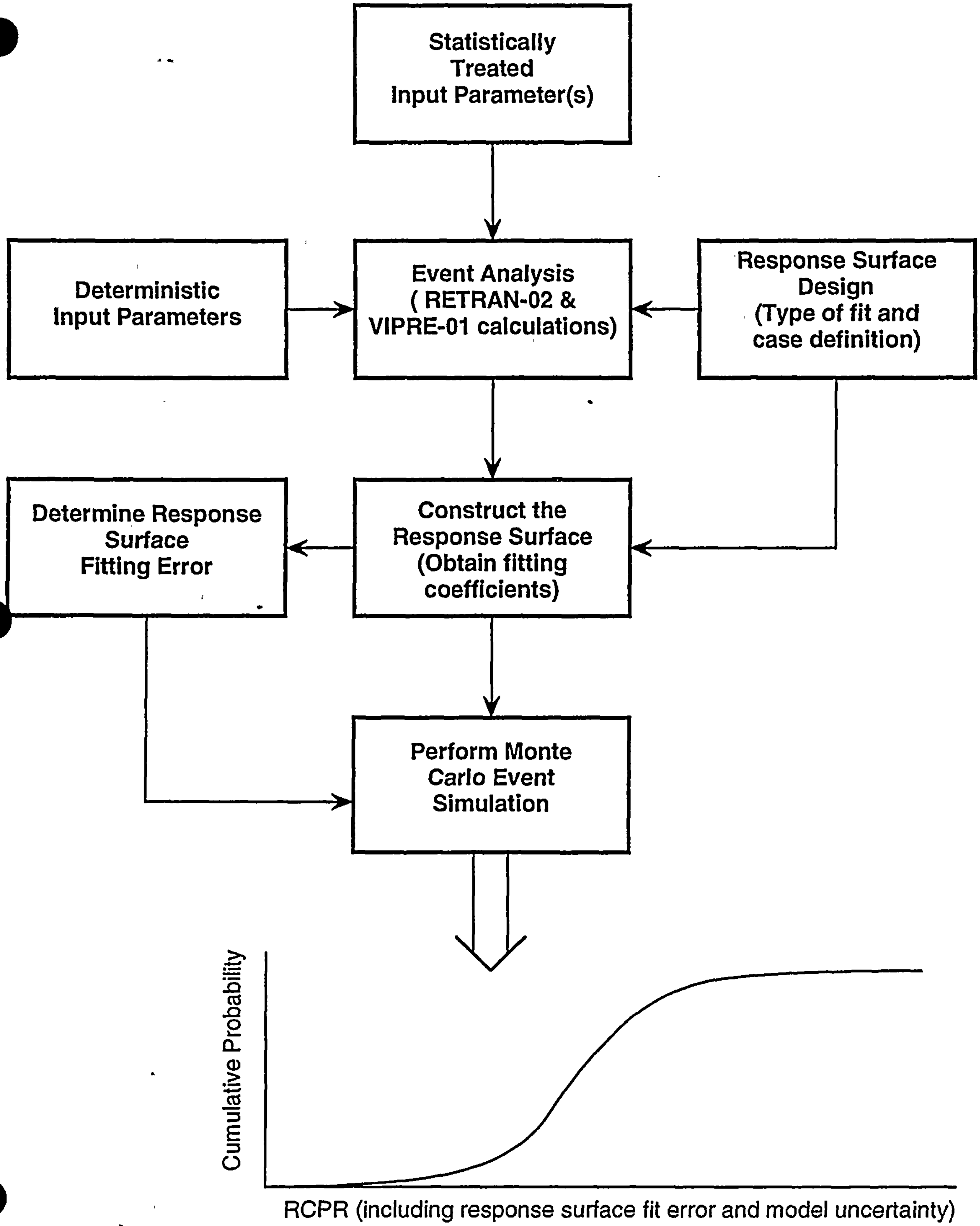
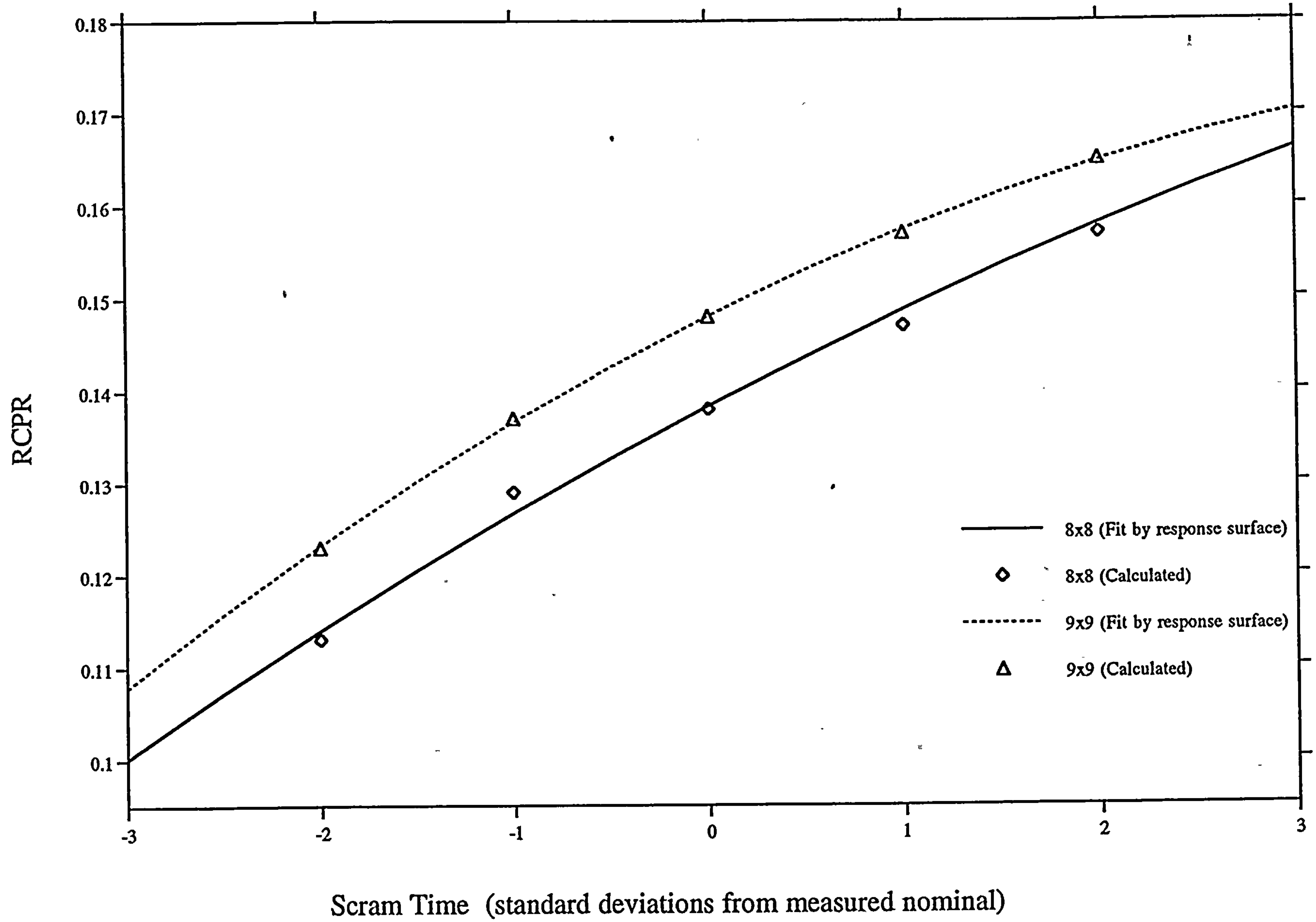


Figure 26-2
RCPR Response Surface



Question 27

For the Feedwater Controller Failure event, there are dramatic differences between the results presented in Rev. 0 and Rev.1. Explain thoroughly these differences (shown on Tables 5.3.7-3 and 4 in Rev. 0 and Tables 5.3.7-4 and 5 in Rev. 1), and provide a complete set of plots comparing these two sets of analyses.

Response

The results presented in Rev. 0 were taken from analysis for Cycle 4 core (all 8x8 fuel). The results presented in Rev. 1 were taken from analysis for Cycle 8 core (mixed 8x8 and 9x9 fuel). A complete set of plots comparing these two sets of analysis are provided in the following pages. The sources for the differences are discussed below.

Improvements and corrections were made to the RETRAN-02 and VIPRE-01 models for WNP-2 after the Cycle 4 simulations were completed. The changes to the RETRAN-02 systems model included: vendor-based gap conductance methodology using the NRC approved RODEX2 code (see Applications Topical Report Revision 1); use of nominal recirculation pump inertia (see Response to Question 17); and mixed core modeling (see Response to Question 16). The initial water level at 104% power for the Cycle 8 analysis was a conservative value of low water level alarm L4 instead of the nominal level that was used in the Cycle 4 simulation. This causes a shift in turbine trip time as can be seen in Figures 27-8 through 27-14. The initial water level for the 47% power cases was set at L4 for both the Cycle 4 and Cycle 8 analyses. Also, the Cycle 4 analysis at 104% power used 100% core flow while the Cycle 8 analysis used 106% core flow.

The changes to the VIPRE-01 hot channel model are the use of vendor-based gap conductance and correction to the heated length calculation for the CPR correlation.

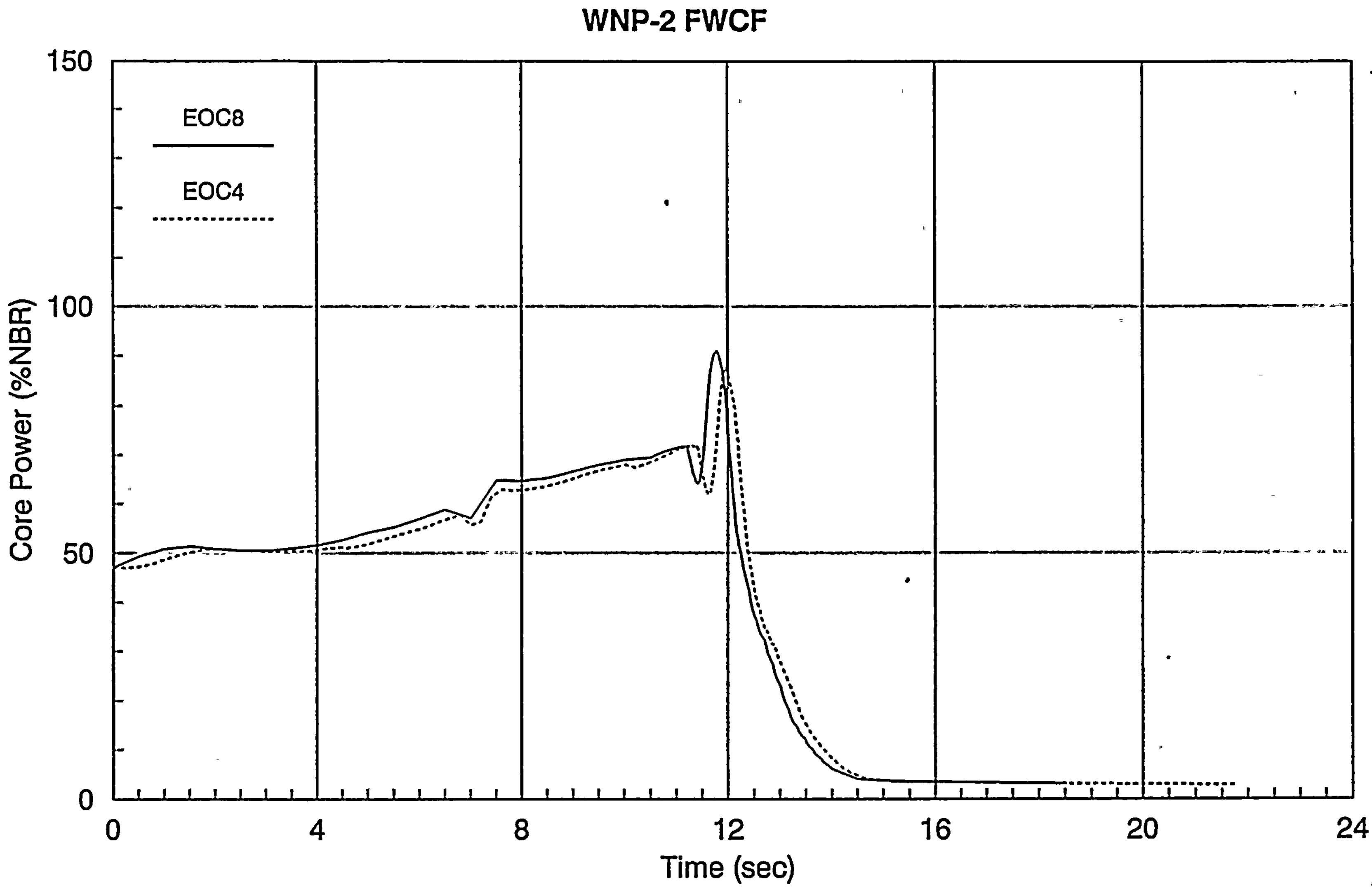


Figure 27-1 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

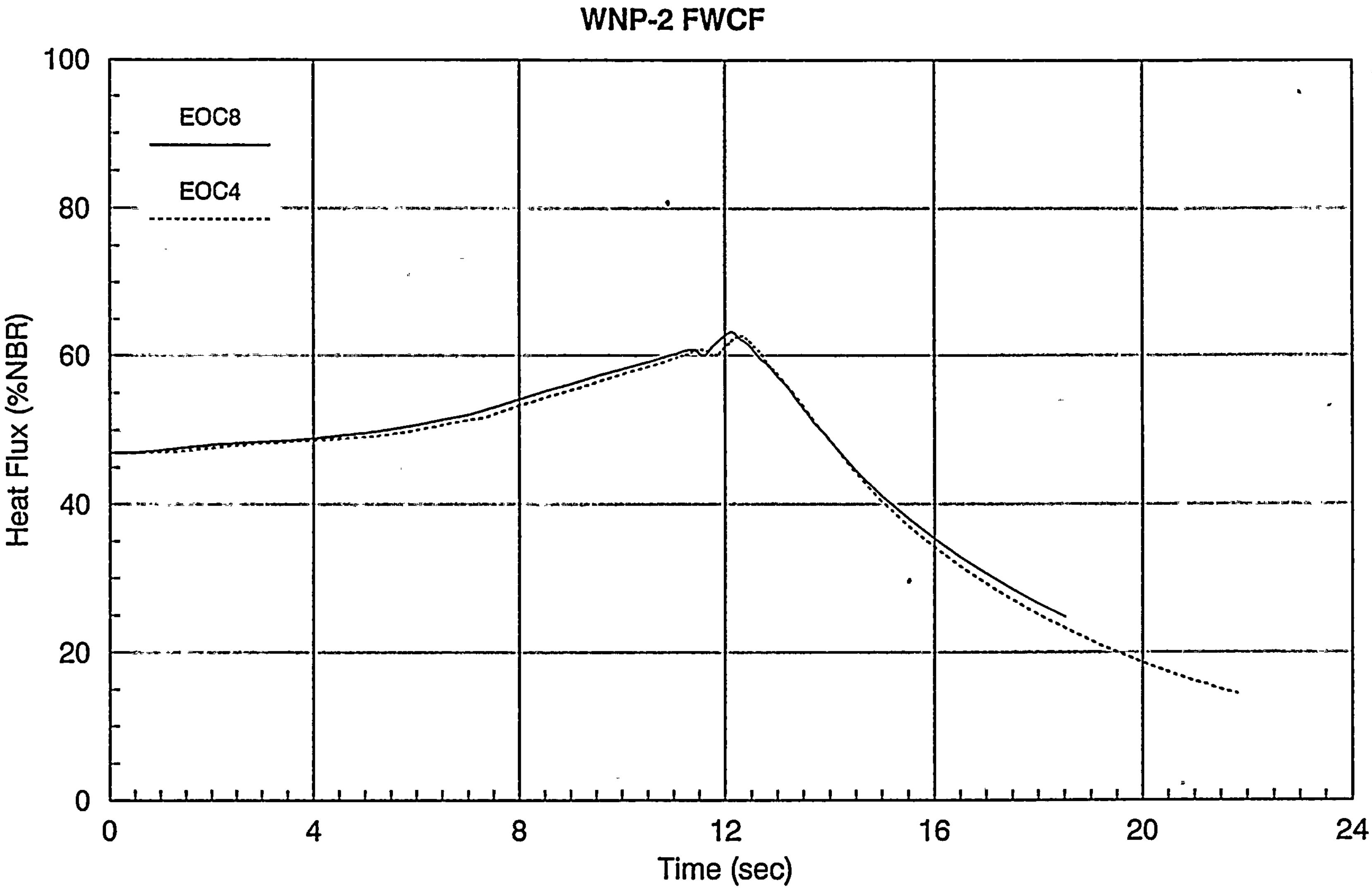


Figure 27-2 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

WNP-2 FWCF

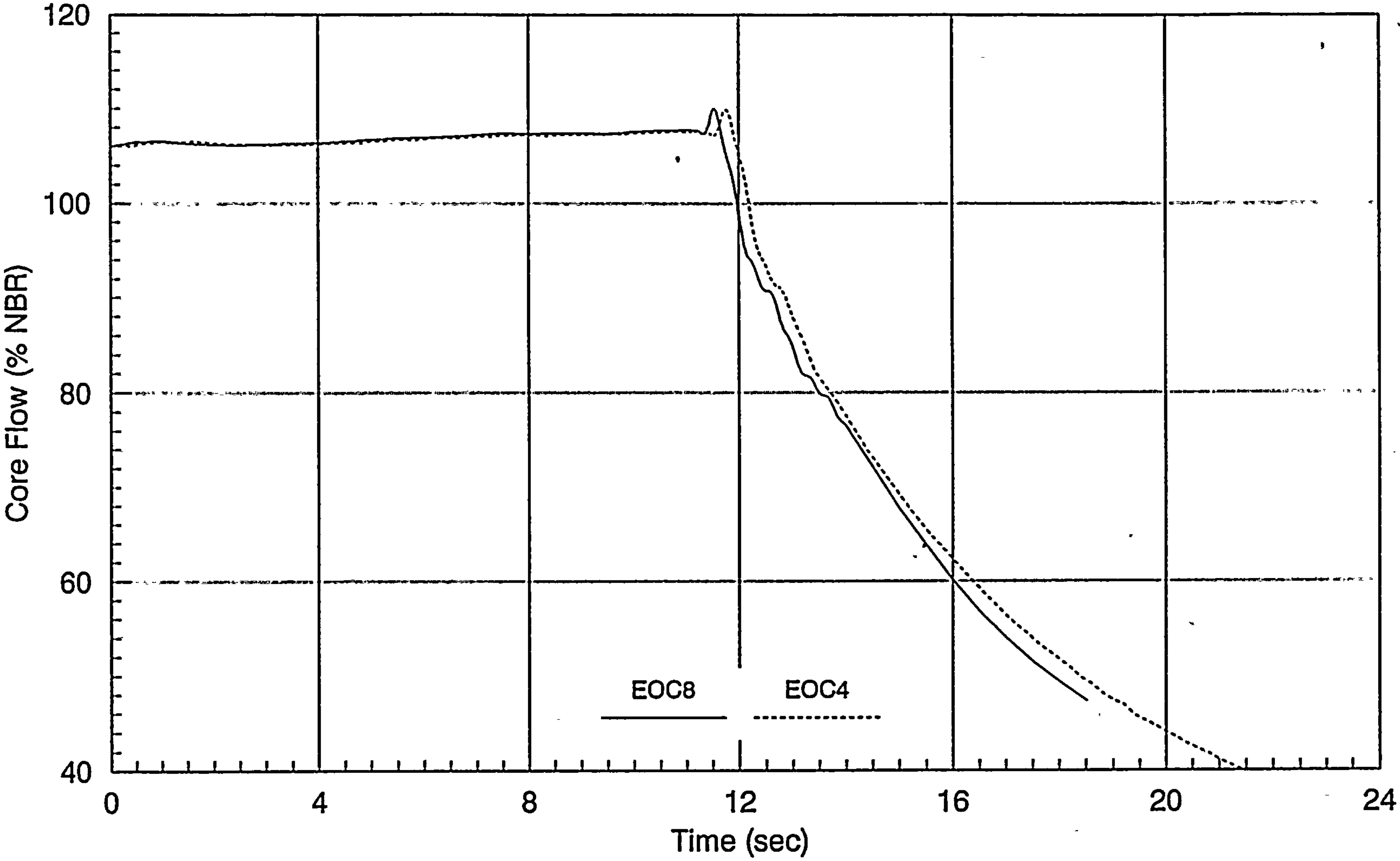


Figure 27-3 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

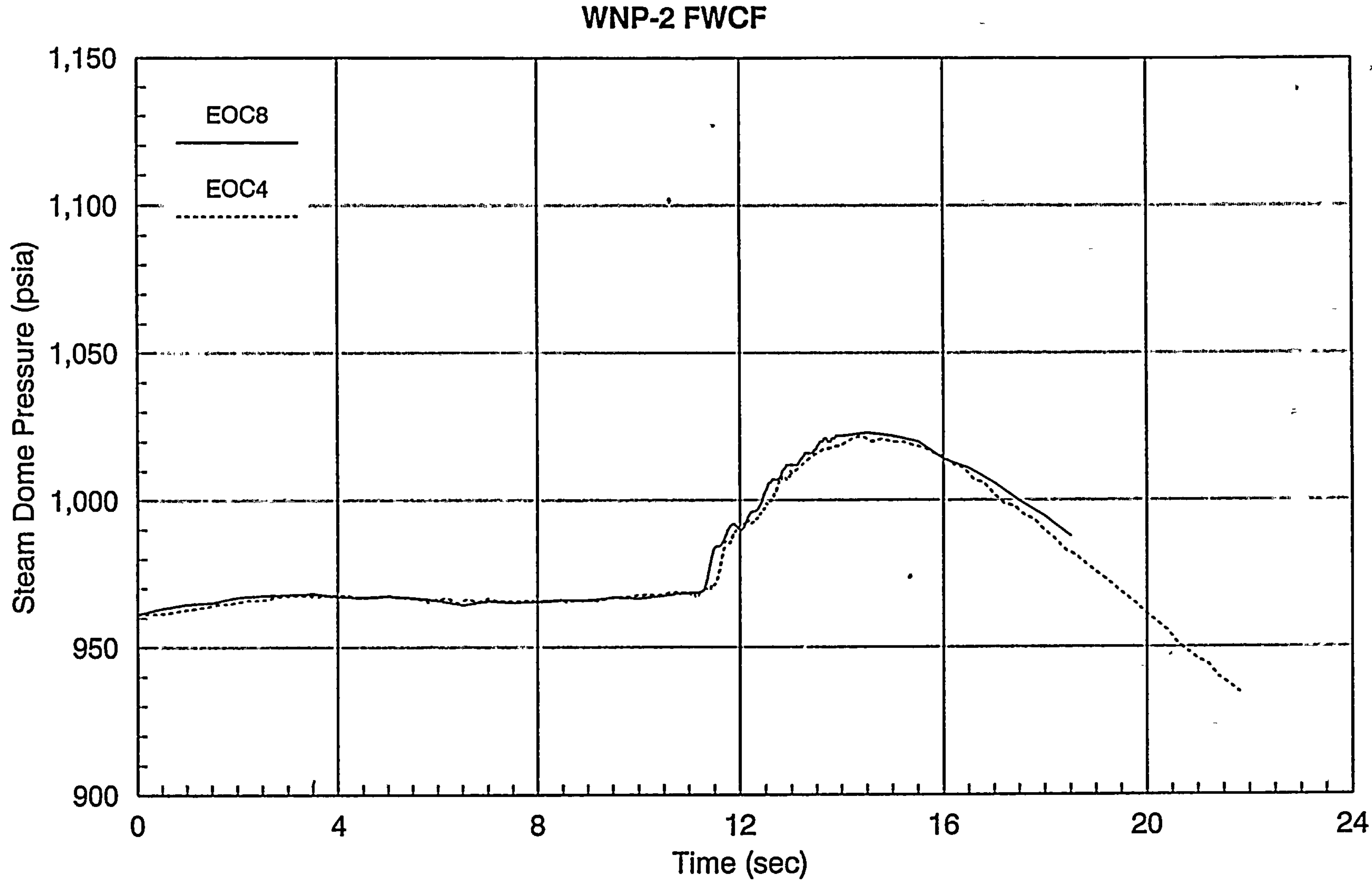


Figure 27-4 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

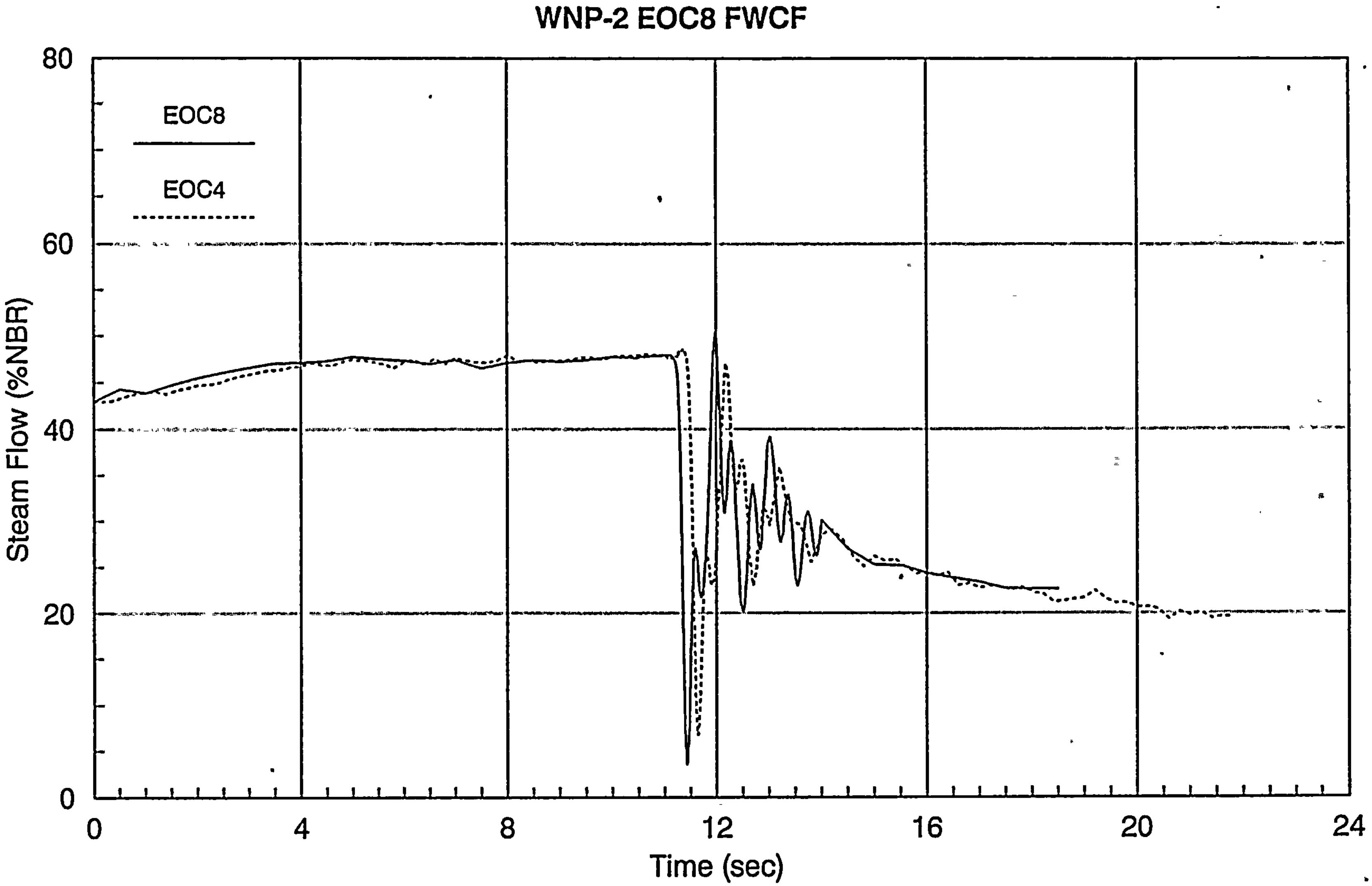


Figure 27-5 FWCF Resuts for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

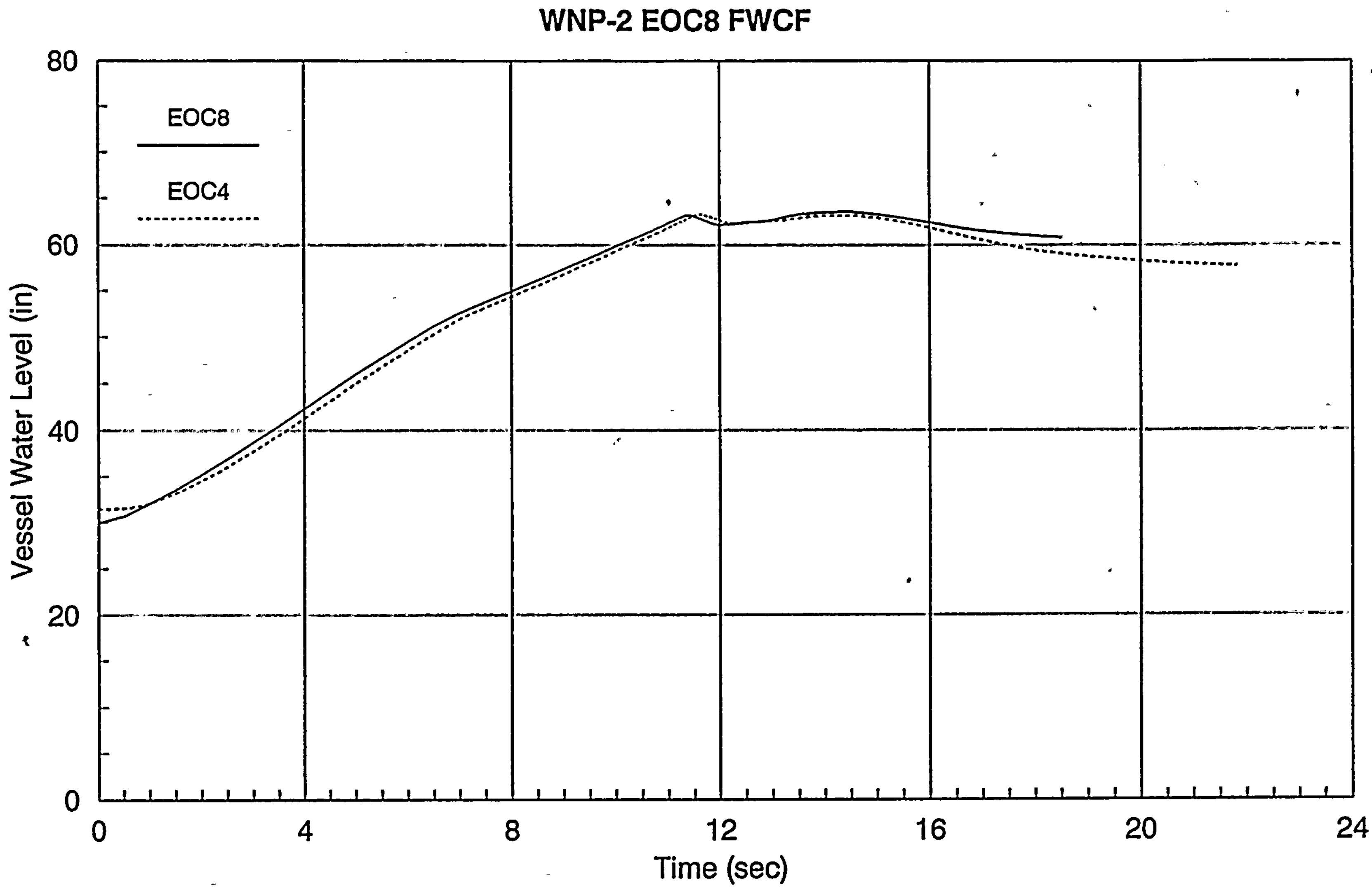


Figure 27-6 FWCF Resuts for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

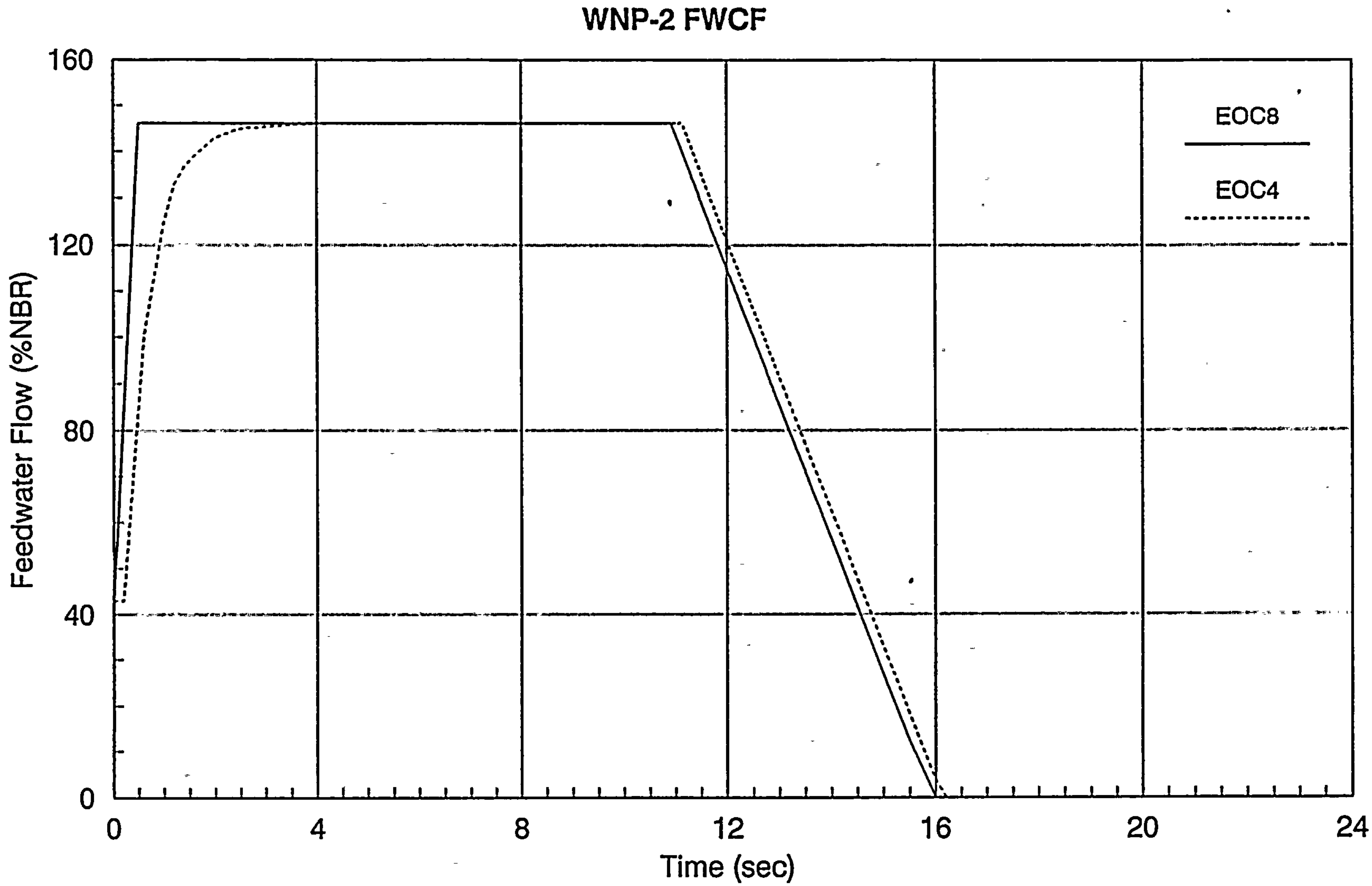


Figure 27-7 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

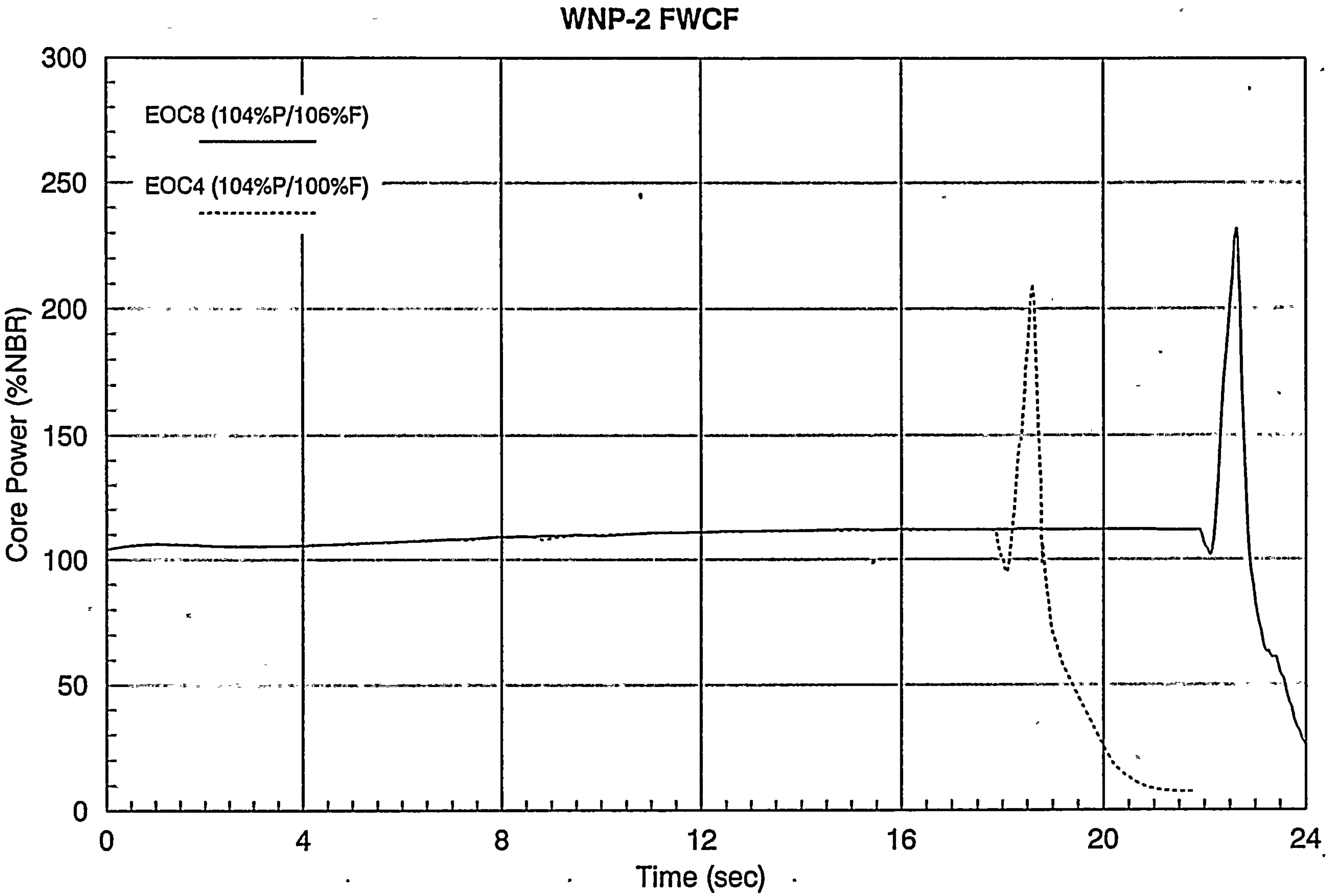


Figure 27-8 FWCF Results, RPT Operable, Tech. Spec. Scram Time

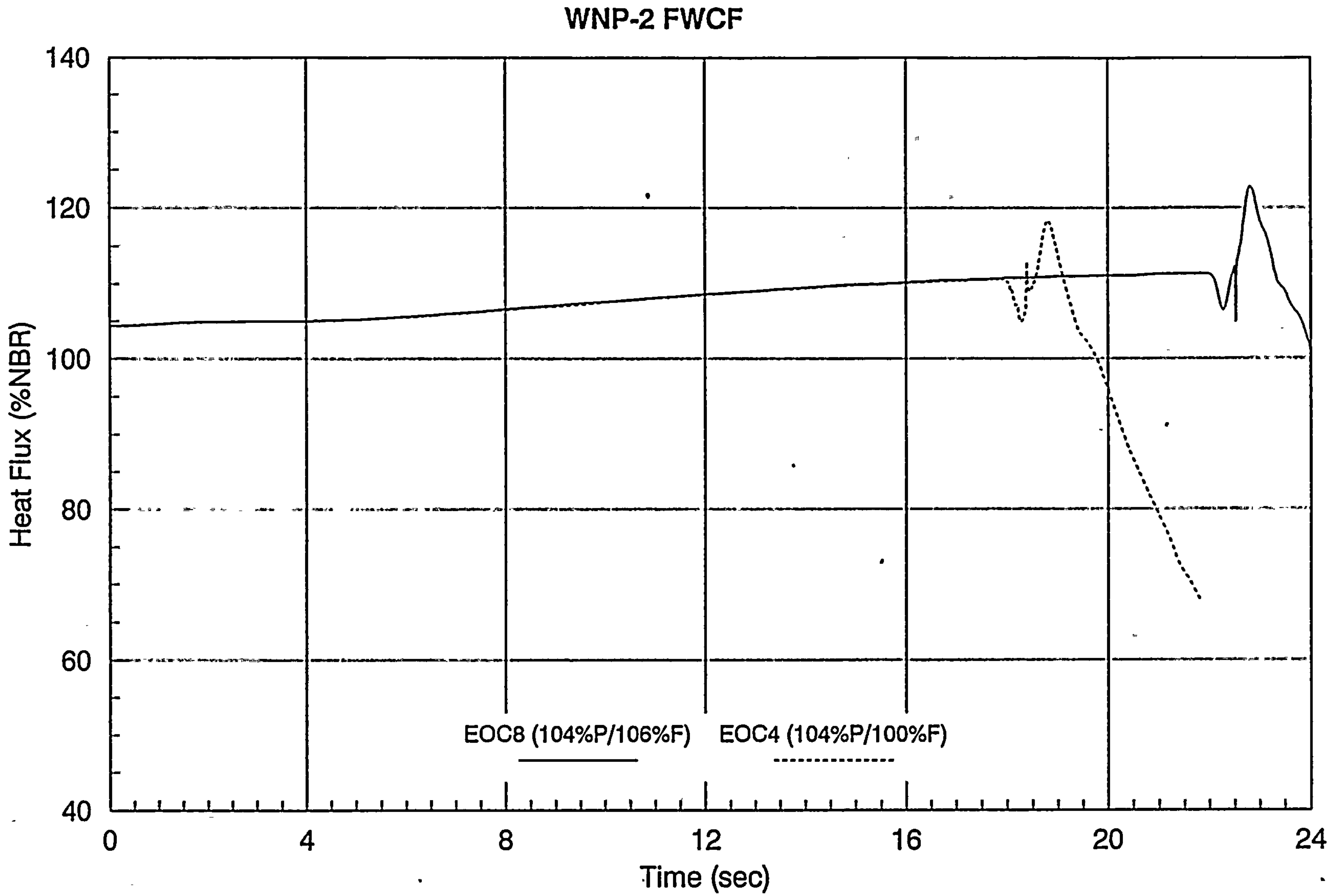


Figure 27-9 FWCF Results, RPT Operable, Tech. Spec. Scram Time

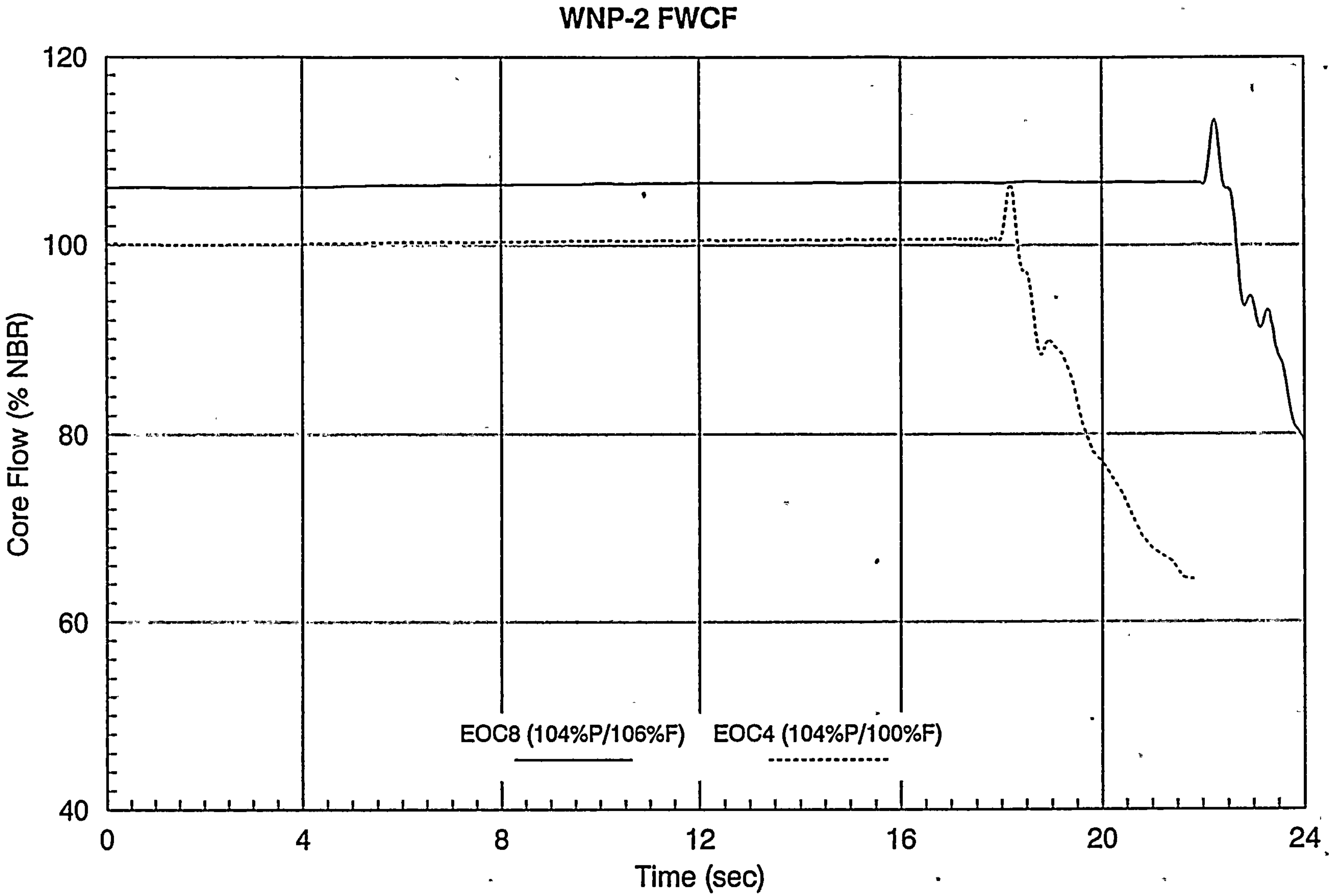


Figure 27-10 FWCF Results, RPT Operable, Tech. Spec. Scram Time

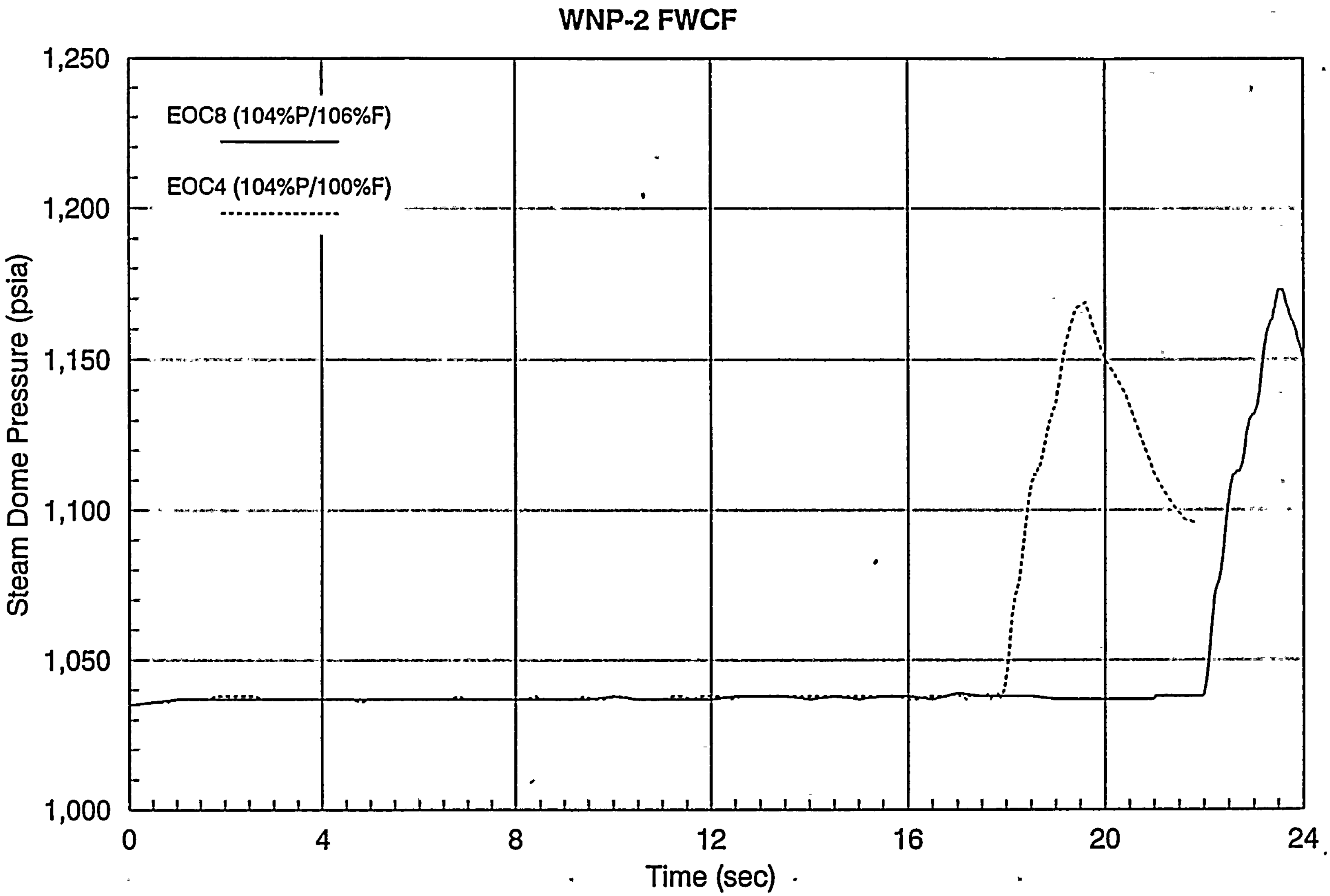


Figure 27-11 FWCF Results, RPT Operable, Tech. Spec. Scram Time

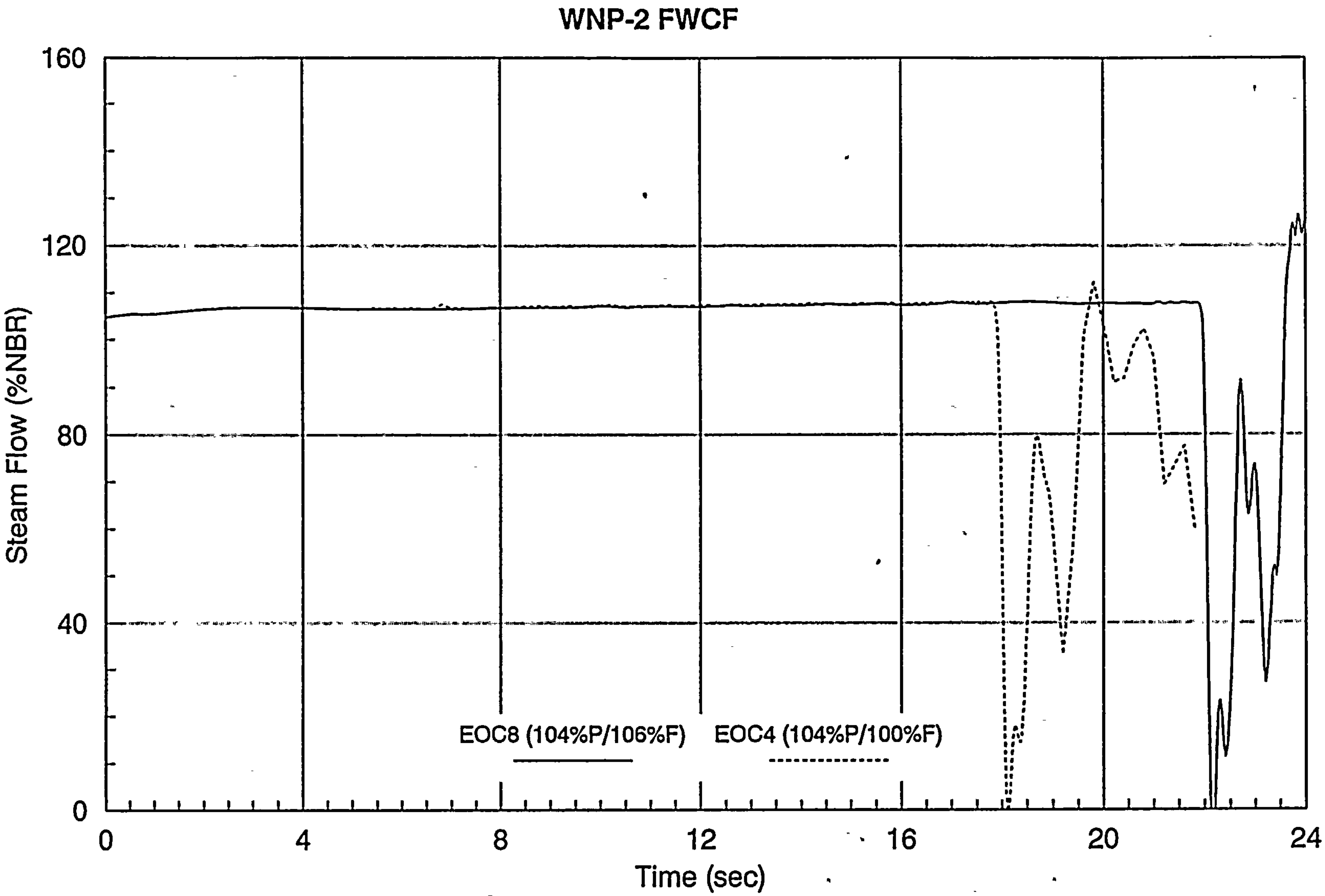


Figure 27-12 FWCF Results, RPT Operable, Tech. Spec. Scram Time

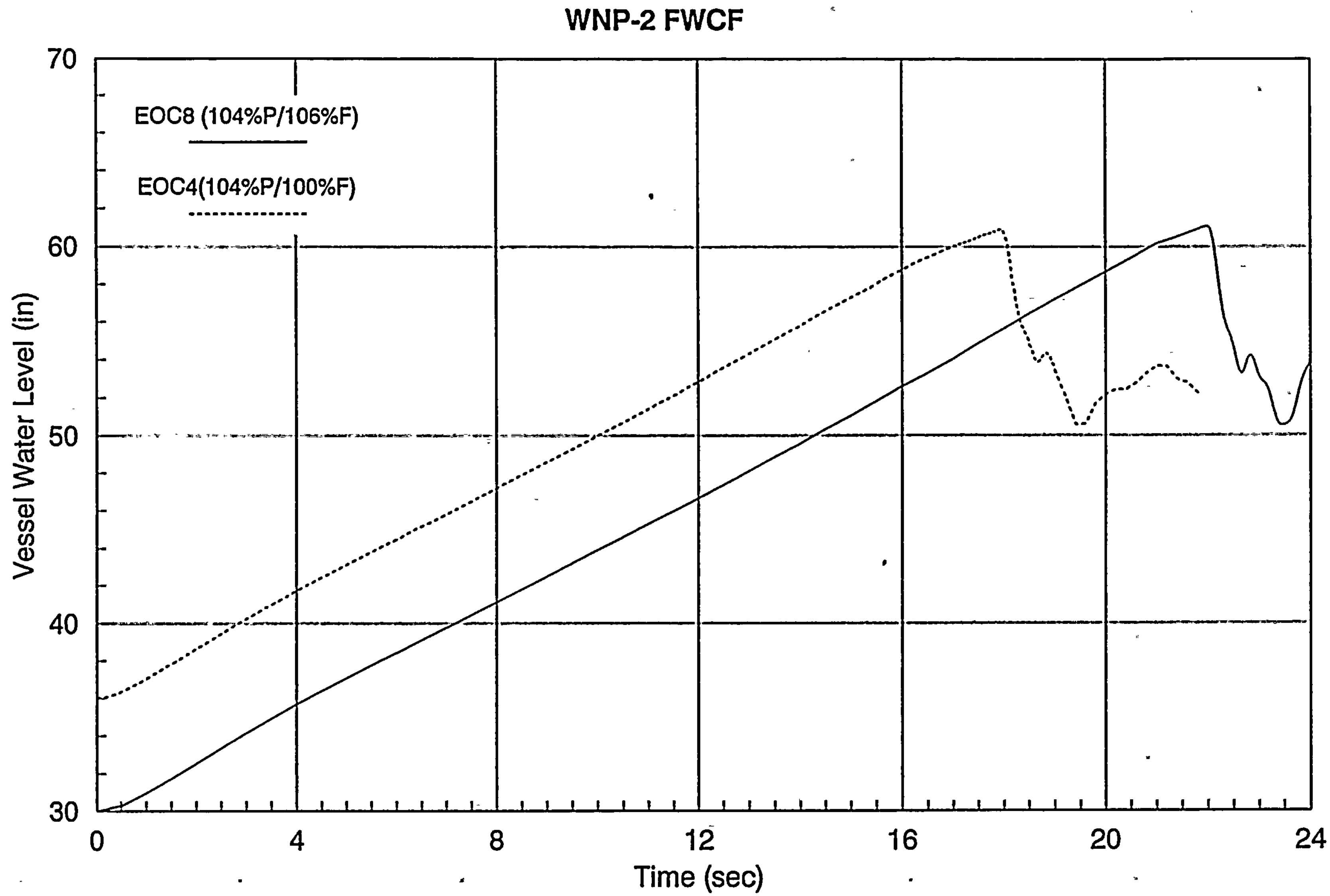


Figure 27-13 FWCF Results, RPT Operable, Tech. Spec. Scram Time

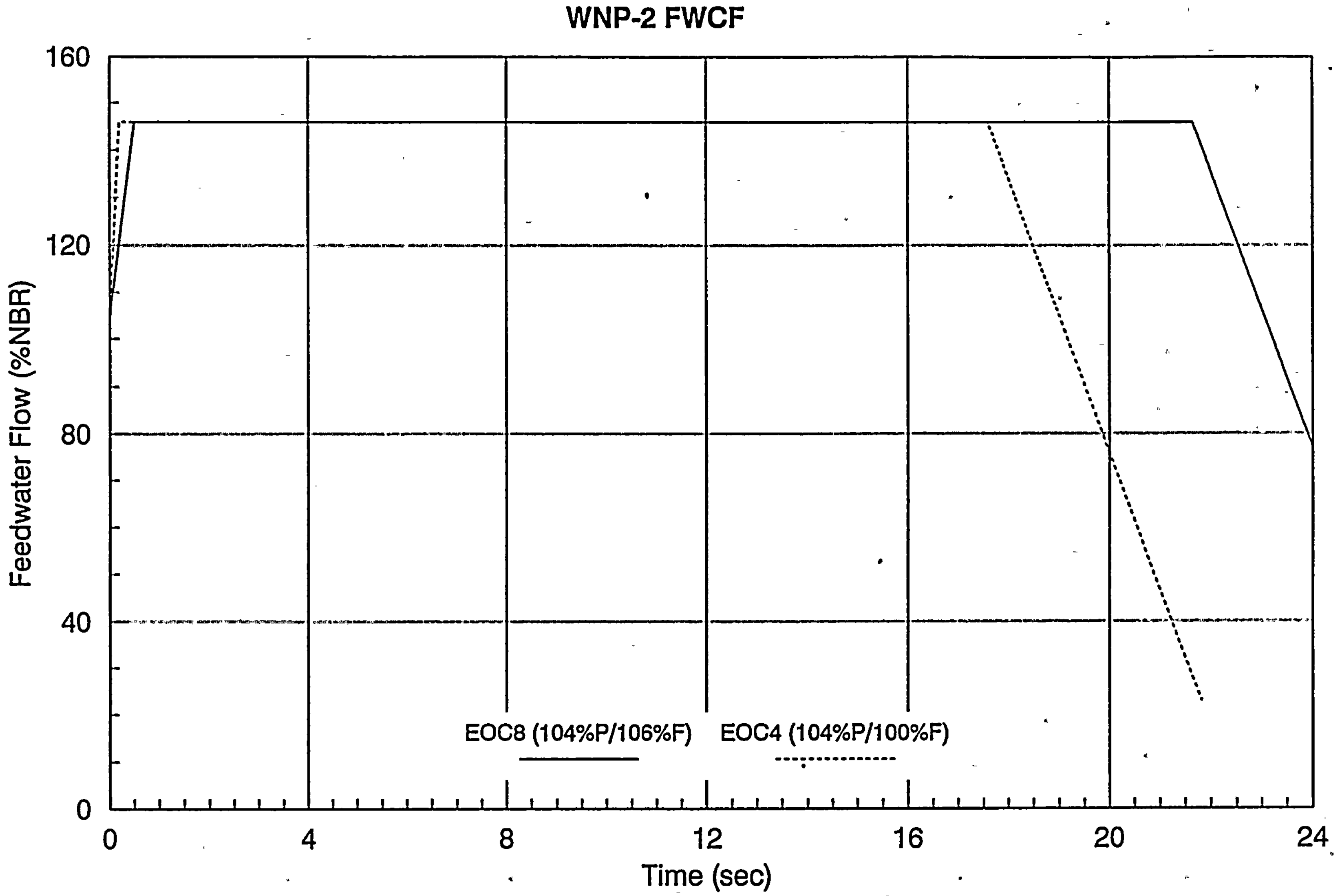


Figure 27-14 FWCF Results, RPT Operable, Tech. Spec. Scram Time

Question 28

Although the event limits for the ASME Overpressure Protection Failure event (Table 5.3.13-2) compare well between the Supply System and ANF results, provide and thoroughly explain further the following comparative information: time of these peaks, transient assumptions and comparative figures for key parameters presented in the submittal.

Response

The results of overpressure protection analysis is provided in the following Table with information on the times of peaks.

	Supply System	SPC
Peak Neutron Flux (% rated)	426 (2.14s)	295 (3.30s)
Peak Heat Flux (% rated)	138 (3.08s)	129 (4.65s)
Peak Reactor Vessel Pressure (psia)	1328 (4.3s)	1303 (5.75s)

The discrepancy in the timing of the peaks is due to the differences in the MSIV closure characteristic. In our Cycle 4 Analysis, we performed a sensitivity study on the MSIV nonlinear closure characteristic. Keeping the total closure time unchanged, SPC analysis used an exponential shape for the MSIV closure characteristic, whereas Supply System used a more realistic S shape curve (Reference 1). This difference in closure characteristic was shown by the Supply System sensitivity study to cause the times of peaks to be delayed by about 1.2 seconds and the peak pressure to be reduced by about 2 psi. Since the limiting parameter for

overpressure protection analysis is the peak vessel pressure, the more limiting closure characteristic (S shape) is selected. The more conservative results of Supply System in comparison to SPC are due to the higher initial rate of void collapse which results in a greater void reactivity feedback in the Supply System model. Initial conditions and transient assumptions described in the Application Topical Report Revision 1, Section 5.3.13.1.1, are identical to those used by SPC. Comparisons of key parameters are presented in Figures 28-1 through 28-5. As can be seen, the transient behaviors are similar except a translation in time ($\approx$ 1.2 seconds).

REFERENCE

1. S.L. Forkner, et al., BWR Transient Analysis Model Utilizing the RETRAN Program, TVA-TR81-01, Tennessee Valley Authority, 1981.

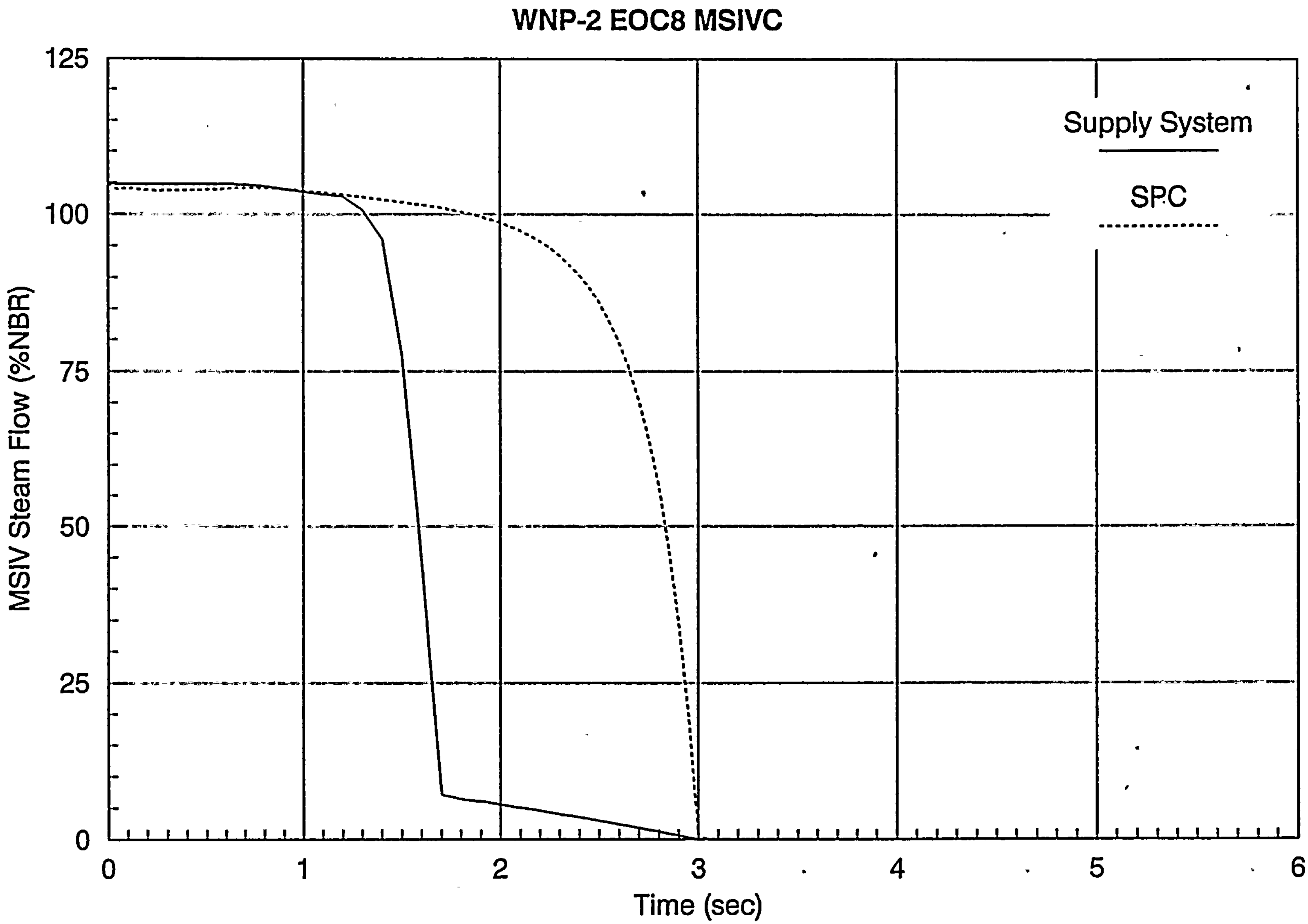


Figure 28-1 MSIV Closure Results, RPT, TSSS

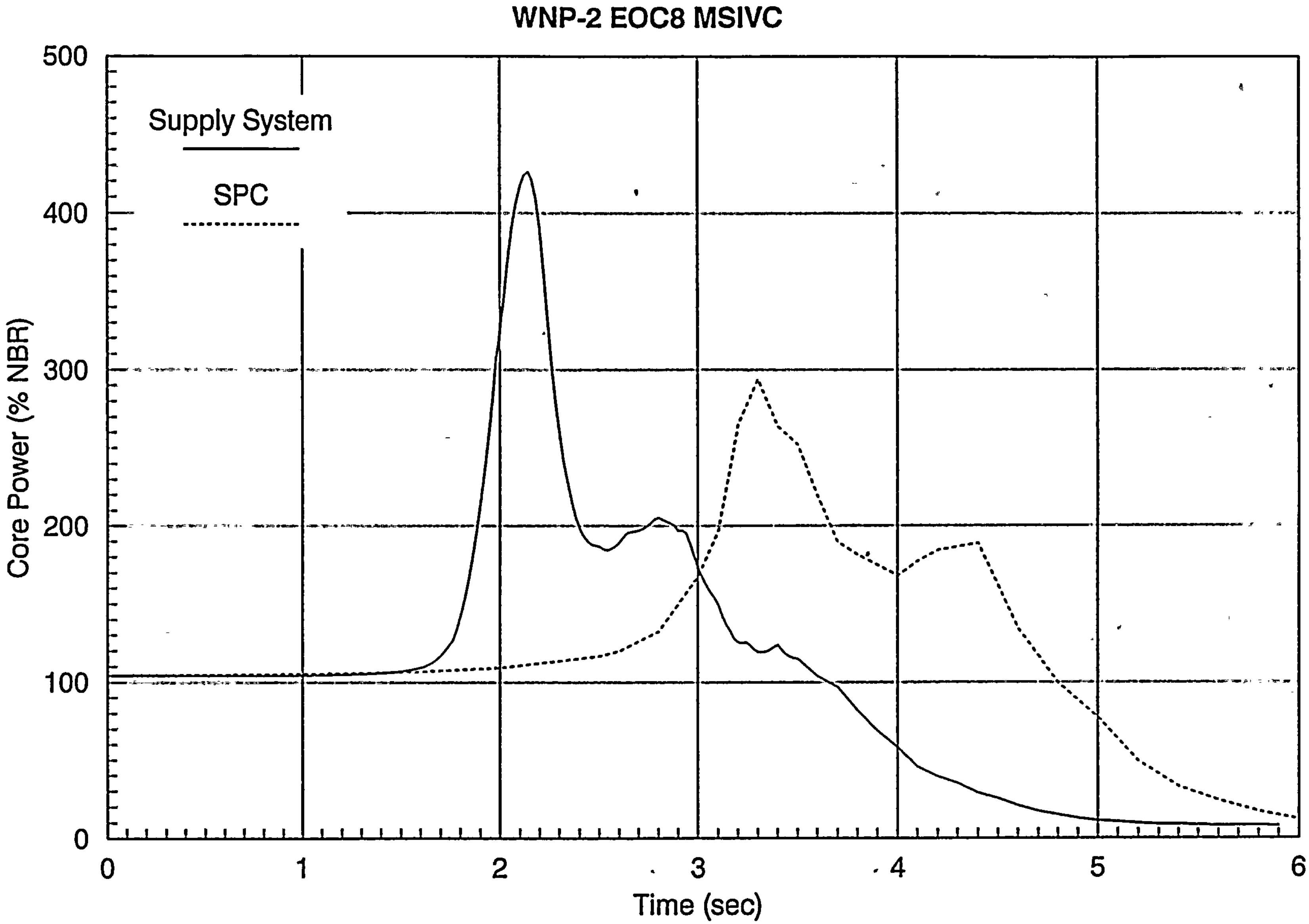


Figure 28-2 MSIV Closure Results, RPT, TSSS

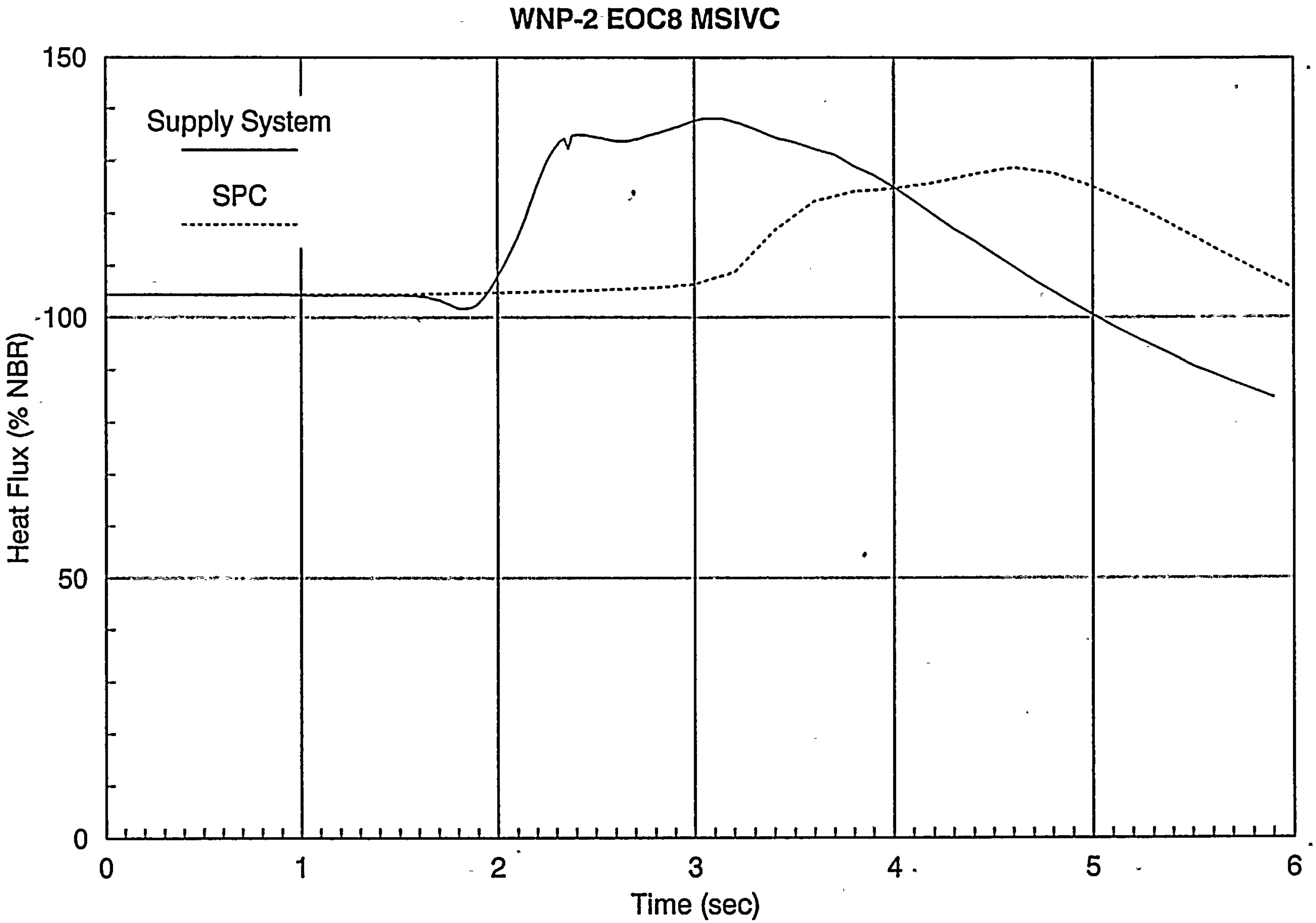


Figure 28-3 MSIV Closure Results, RPT, TSSS

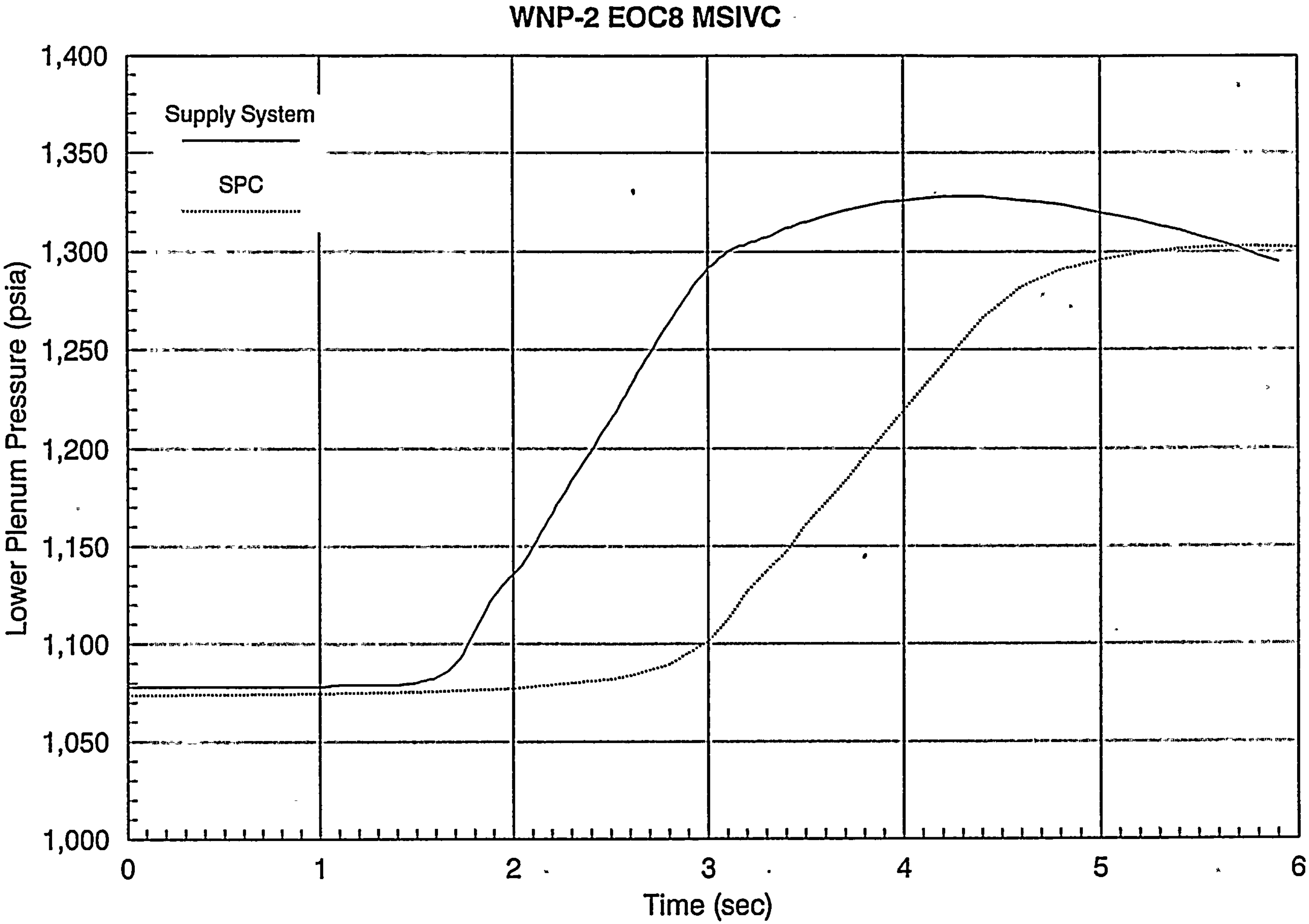


Figure 28-4 MSIV Closure Results, RPT, TSSS

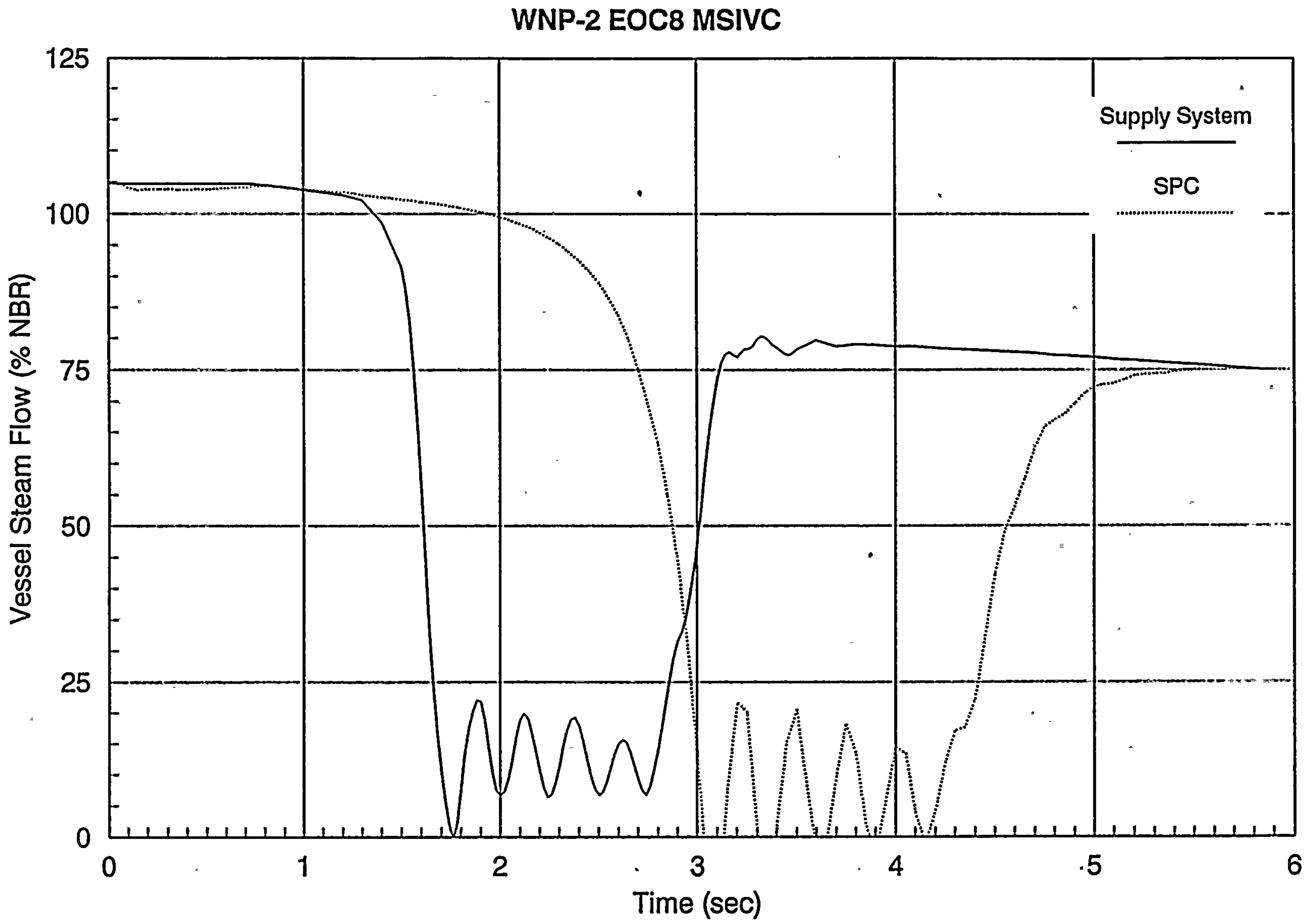


Figure 28-5 MSIV Clousure Results, RPT, TSSS

TECHNICAL EVALUATION REPORT OPEN ISSUES

TER Open Issue 1

Technical Evaluation Report (TER) p.6 recommends that Supply System retain the option to use finer core noding for cases where fine structures of core parameter are necessary.

Response

As stated in the TER (Reference 1), the Supply System has performed a sensitivity study in which the core noding was increased from a 12-node core to a 24-node core for the Power Ascension Test No. 27 (Load Rejection Test). It was found that the 12-node model yielded essentially the same results as the 24-node model. However, the Supply System will retain the option to use finer core noding for cases where fine structures of core parameters are determined necessary.

REFERENCE

1. Letter, J. Clifford (NRC) to J.V. Parrish, BWR Transient Analysis Model (TAC Nos. M77048 and M81723), October 25, 1993

TER Open Issue 2

TER p. 6 states that care needs to be taken when performing licensing type calculations to assure conservative predictions. It further states that the Supply System should identify the transients for which tracking of the voids in the core and reactivity are important and justify the use of the coarse nodalization for those cases.

Response

As stated in the response to TER Open Issue 1, the Supply System has performed a sensitivity study in which the core noding was increased from a 12-node core to a 24-node core for the Power Ascension Test No. 27 (Load Rejection Test) and found that the 12-node model yielded essentially the same results as the 24-node model. In addition, for the much more severe transients of Peach Bottom Turbine Trip Test, it was found that the Supply System best-estimate 12-node RETRAN-02 model yielded more conservative power histories than the measured data for all three tests.

TER Open Issue 3

TER p. 7 states that the Supply System should provide additional justification for conditions with single-phase fluid in the upper downcomer, if such conditions occur.

Response

The limiting Anticipated Operational Occurrence transients analyzed using the licensing basis model have steam-liquid interface in the upper downcomer region where the non-equilibrium model is used. For other evaluations in which there is a single-phase fluid in the upper downcomer region, other modeling approaches will be taken to appropriately simulate these events.

TER Open Issue 4

TER p. 8, under the jet pump model heading, states that the Supply System is required by the RETRAN-02 SER to provide justification of its use for flow prediction in transients for which a negative flow is expected, and to quantify and justify its impact on the predicted transient MCPR and the determination of conservative plant operating limits.

Response

The justification for using RETRAN-02 for a transient where a reversed jet pump flow exists is given in response to Question 23.

TER Open Issue 5

TER p. 10 states that the Supply System should identify the transients for which the feedwater controller is relied upon and faster acting response is necessary, and that information should be examined during the review of the Applications Topical to assure that low prediction of water level is conservative in terms of licensing applications.

Response

See the response to Question 20. Among the potentially limiting events analyzed with the RETRAN-02 model, only Feedwater Controller Failure (FWCF) is directly impacted by the water level calculations. Generator Load Rejection without Bypass (LRNB) and Main Steam Line Isolation Valve Closure (MSIVC) are not sensitive to the water level predictions. As stated in the TER, the prediction of lower water level is more conservative for the FWCF transient.

TER Open Issue 6

TER p. 10 states that the Supply System uses a bounding value for the recirculation pump inertia in the licensing basis model.

Response

See the response to Question 17. The current licensing model uses the nominal pump inertia for both recirculation pumps.

TER Open Issue 7

TER p. 12 recommends that mixed core effect analysis be addressed during the review of the Applications Topical Report.

Response

Mixed core effects are addressed in the responses to Questions 13, 14 and 16.

TER Open Issue 8

TER p. 12 requests that the adequacy of plant key parameters for analysis be determined during the review of the applications topical report.

Response

See the responses to Questions 4 and 17.

TER Open Issue 9

TER p. 13 recommends that the adequacy of its (slip algebraic slip model) use should be reviewed during the review of the Applications Topical Report.

Response

The algebraic slip model is addressed in the response to Question 11.

TER Open Issue 10

TER p. 13 requests that before any licensing analysis is accepted, the Supply System submit (i) statistical analysis; (ii) justification of its licensing conditions and safety margins; and (iii) comparative analysis of the standard NRC problem (a Turbine Trip Without Bypass transient).

Response

For Item (i), please see the responses to Questions 4, 5, 6, 7 and 26. For Item (ii), please see the responses to Questions 17 and 18. For Item (iii), comparative analysis of the standard NRC problem (a Turbine Trip without Bypass Transient) was submitted to the NRC along with our responses to the second round of questions on Topical Report WPPSS-FTS-129 (Letter, GC Sorensen to NRC, "Response to Request for Additional Information Regarding Topical Report WPPSS-FTS-129, BWR Transient Analysis Model (TAC No. 77084)", dated March 5, 1992).

TER Open Issue 11

TER p. 14 states that detailed discussion and justification of the use of jet pump model with flow reversal should be provided in the context of existence of a core safety limit margin.

Response

See the response to Question 23. The Safety Limit MCPR has an adder for single loop operation which accounts for core flow uncertainty.

TER Open Issue 12

TER p. 14 states that the mixed core effect analysis be reviewed as part of the reload methodology during the review of the Applications Topical Report.

Response

See the responses to Questions 13, 14, and 16.