APPLICATIONS TOPICAL REPORT FOR BWR DESIGN AND ANALYSIS

WPPSS-FTS-131

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DISCLAIMER

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ABSTRACT

This report documents the ability of the Supply System to perform reload analyses that ensure successful reload fuel designs and core configurations. These analyses also ensure compliance with all regulatory requirements and conformance with industry practice.

The reload analysis begins with an energy utilization plan and a reference core design for the next reactor cycle. An integrated set of computer codes and a range of analysis techniques, such as the statistical combination of uncertainties, make it possible to evaluate limiting events. These evaluations provide the bases for any changes in operating limits and technical specifications and ensure that WNP-2 will operate safely with each fuel reload and each new core configuration. The Supply System will document the results of each reload analysis in a reload summary report.

SECTION 1.0
INTRODUCTION

The Washington Public Power Supply System (WPPSS or Supply System) operates the Washington Nuclear Project No. 2 (WNP-2). Designed by General Electric, WNP-2 is a BWR/5 boiling water reactor (BWR) with a licensed core thermal power of 3323 MWt. The Supply System has developed a methodology for the reactor safety analysis. This report describes the application of that methodology to the fuel reload design.

1.1 RELOAD ANALYSIS OVERVIEW

The reload analysis begins with development of the following:

- an energy utilization plan that establishes operational goals for the reload
- a prediction of the end-of-cycle core condition for the current operating cycle
- a reference fuel cycle.

These sources provide information for a tentative fuel design and reference loading pattern, which are then evaluated with the lattice physics codes and a three-dimensional core simulator code.

After the fuel design and reference loading pattern have been revised as necessary, a safety analysis evaluates potentially limiting events. These events have been divided into three categories that reflect the probability of occurrence and analysis requirements:

- normal operation and anticipated operational occurrences
- accidents
- special events.

Screening identifies the potentially limiting events in each of the three categories. These events are then re-evaluated during each reload analysis.

Analyses of limiting events provide the bases for any changes in core operating limits or technical specifications. Lattice physics methods, the three-dimensional simulator code, and transient analysis are used for evaluating all three categories of events. Event analyses for rod drop accidents and loss of coolant accidents (LOCA) rely on the fuel vendor methodology.

Cycle-specific reload summary reports document the results of reload analyses, which provide the basis for any changes in core operating limits or technical specifications. Figure 1-1 provides an overview of the reload analysis process.

1.2 METHODOLOGY IMPLEMENTATION AND UPDATING

Implementation of the reload analysis methodology will require amending the technical specifications so they reference this topical report instead of the various fuel vendor topical reports. This reload analysis methodology will also be updated as required to reflect changes in design and technology, and all Supply System documents will reflect these changes.

Approved copies of this report will be issued in loose leaf binders. Revisions will be issued as page changes. Page change records will provide traceability to previous versions and identify firmly which methodology was in place when a specific analysis was done. Furthermore, every page of the report will include the date of the applicable Safety Evaluation Report (SER).

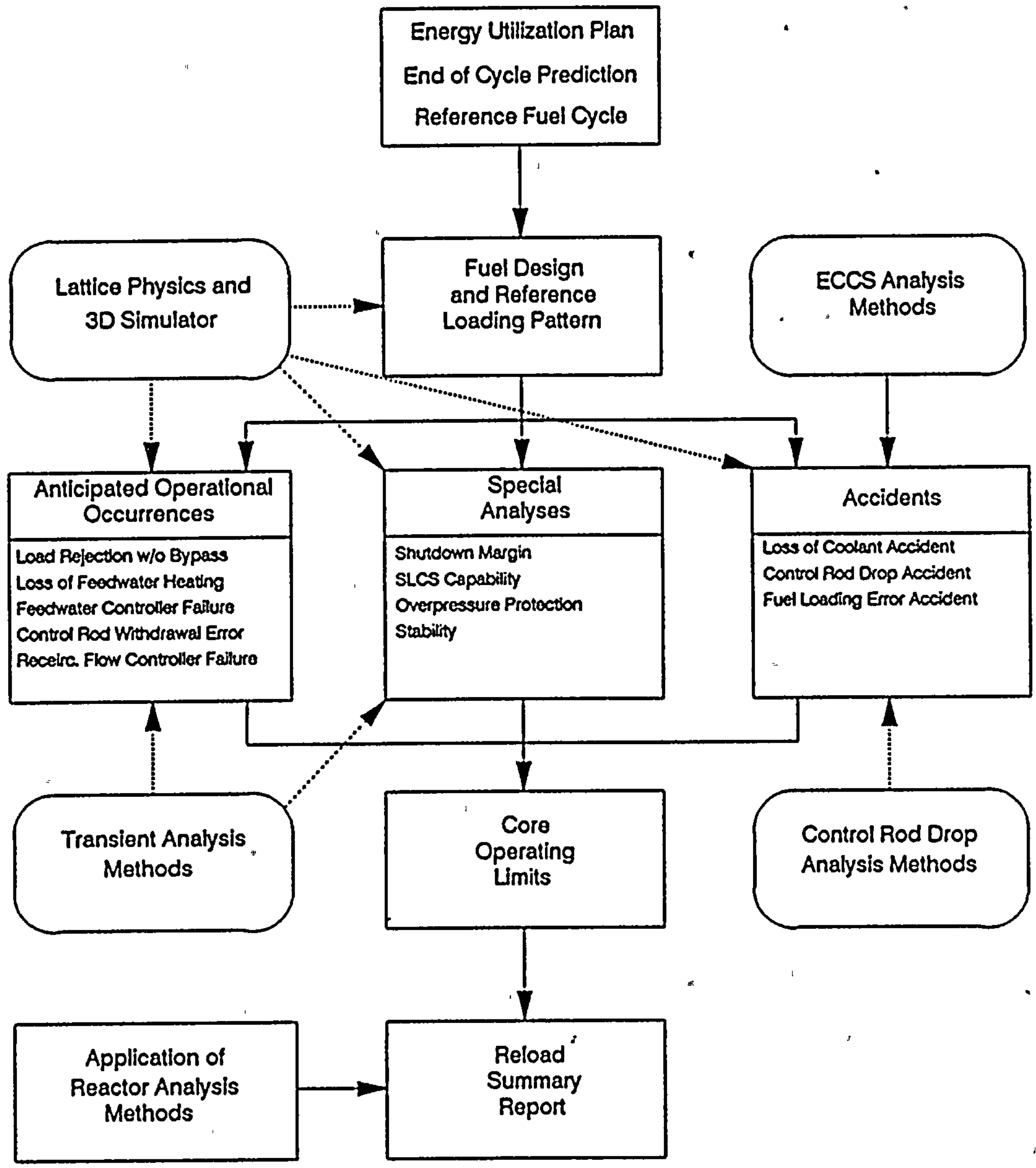

Figure 1-1. Reload Analysis Overview

SECTION 2.0

COMPUTER CODES AND ANALYSIS TECHNIQUES

The reload analysis ensures that the reload fuel design and core configuration satisfy regulatory requirements. To meet these requirements, the reload analysis draws on a number of computer codes and analytical techniques for the design analysis and safety analysis. Uncertainties in the results may be evaluated using either a deterministic approach or the statistical combination of uncertainties (SCU) methods.

Together the design analysis and safety analysis accomplish the following:

- establish an operating limit minimum critical power ratio (MCPR) that ensures the fuel cladding integrity limit will not be exceeded as a result of an anticipated operational occurrence or a fuel loading error
- demonstrate that the protection against fuel failure (PAFF) limits will not be exceeded as a result of an anticipated operational occurrence
- establish maximum average planar linear heat generation rate (MAPLHGR) limits that ensure the consequences of a LOCA satisfy the requirements of 10CFR50.46
- demonstrate that the reload core design has sufficient shutdown margin
- demonstrate that the core can be shut down without the control rods
- demonstrate that the consequences of a control rod drop accident fall within regulatory bounds
- demonstrate that the results of the Code overpressure protection analysis meet the requirements of 10CFR50.55a.

2.1 CODE DESCRIPTIONS

The Supply System methodology uses a suite of computer codes and associated models and code inputs to analyze the limiting events associated with the design and safety analyses. The Supply System codes consolidate core physics, system thermal-hydraulics, and core thermal-hydraulics into an integrated system. These codes are comparable to those used by nuclear steam supply system (NSSS) and reload fuel suppliers, and have been extensively validated by comparison to plant data, experimental test facility data, and benchmark analytical calculations. The methodology draws particularly heavily on the Reactor Analysis Support Package (RASP) system of computer codes developed by the Electric Power Research Institute (EPRI) [Reference 1].

Several utilities use RASP for a range of applications, including

- fuel cycle management
- core design
- plant reload licensing support
- development of analytical bases to support changes in plant design and configuration, protection system setpoints, and technical specifications.

The Supply System supplements the RASP codes with a number of special purpose codes. These codes have been implemented by the Supply System and are integrated with specific facets of the fuel supplier methodology. The Rasp codes and special purpose codes can also be used to perform other analyses that support operations.

Figure 2-1 shows the computer codes and code sequence. All computer codes are maintained and controlled by the NRC-approved Supply System quality assurance program (see letter D.F. Kirsch of NRC to WPPSS, "Operational Quality Assurance Program Description," WPPSS-QA-004, Rev. II, June 8, 1987).

2.1.1 Reactor Physics Codes

The reactor physics codes model steady-state core conditions; simulate slow transients that can be analyzed with quasi-steady-state methods; and provide neutronics input for dynamic analyses. These codes include

- o MICBURN-E, a pin cell code to treat gadolinia isotopic depletion
- o CASMO-2E, a lattice physics code
- o SIMULATE-E, a three-dimensional core simulator code
- o NORGE-B, a linking code for transferring data from CASMO-2E to SIMULATE-E and to the core neutronics linkage code, SIMTRAN-E.

MICBURN-E and SIMULATE-E were developed by EPRI [References 2, 3]. NORGE-B was developed by EPRI and modified by the Supply System [Reference 4]. CASMO-2E was developed by Studsvik Energiteknik AB.

2.1.1.1 MICBURN-E

MICBURN-E is a multigroup, pin-cell physics code for analysis of light water reactor fuel rods with homogeneous mixtures of uranium (UO_2) and gadolinia (Gd_2O_3). MICBURN-E uses integral transport theory to calculate neutron flux distributions and eigenvalues; it employs a time-dependent analysis method to calculate fuel rod isotopic distributions as a function of burnup.

In the reload analysis, MICBURN-E generates burnable absorber data for input to CASMO-2E. The resulting files contain effective microscopic absorption cross sections tabulated as a function of burnup for a pseudo-burnable absorber nuclide equivalent to the constituent absorber nuclides of the gadolinia.

2.1.1.2 CASMO-2E

CASMO-2E is a multigroup, two-dimensional transport theory code used for burnup calculations on fuel assemblies or simple pin cells. This code uses transmission probabilities to solve the neutron transport equation within a two-dimensional representation of the fuel assembly.

In BWR applications, CASMO-2E treats cylindrical fuel rods of varying composition in a square pitch array, with allowance for fuel rods loaded with gadolinia. It also handles fuel assembly channels, water gaps between fuel assemblies, incore instrumentation, and cruciform control rods.

Input data to CASMO-2E includes the following:

- composition data for the fuel and structure
- burnable absorber data produced by MICBURN-E
- assembly and control rod dimensions
- internal fuel rod arrangement
- average fuel temperature from ESCORE analyses.
- power density

CASMO-2E performs the following sequence of calculations:

- a resonance calculation
- a pin cell calculation for each type of fuel rod in the assembly
- a one-dimensional spectrum calculation
- a two-dimensional power distribution and k_∞ calculation
- a calculation of assembly-averaged, two-group cross sections and kinetics parameters.

For each lattice design, CASMO-2E also calculates neutronics data as a function of the following BWR parameters:

- o exposure history
- o exposure-averaged relative moderator density
- o instantaneous relative moderator density
- o control rod presence
- o control rod history
- o fuel temperature
- o moderator temperature
- o xenon concentration
- o samarium concentration
- o boron concentration.

CASMO-2E provides the cross section data used by SIMULATE-E in the core analysis and the inverse neutron velocities and delayed neutron fraction used by SIMTRAN-E. In addition, CASMO-2E produces the local peaking factor distribution and incore detector response. CASMO-2E has qualified for use in the reload analysis [Reference 5].

2.1.1.3 NORGE-B

NORGE-B is a data handling and processing code used for transferring data from CASMO-2E to SIMULATE-E and through SIMTRAN-E to the transient analysis codes. NORGE-B provides SIMULATE-E with the following:

- o partial cross sections in the form of two-dimensional interpolating tables and polynomial fits
- o additional polynomial fits for isotopic fission yields and neutrons per fission.

NORGE-B provides SIMTRAN-E with the following:

- o two-group inverse neutron velocities
- o the total effective delayed neutron yield.

2.1.1.4 SIMULATE-E

SIMULATE-E is a coarse-mesh, three-dimensional, nodal-diffusion code used for steady-state core analysis. It models the reactor core as a matrix of neutronically coupled nodes, each a six-inch axial segment of a fuel assembly. The core analyses use cubic nodes. For each node, SIMULATE-E uses two-group macroscopic cross section data to solve the coarse-mesh diffusion theory equations in one energy group. Fast and thermal neutron flux distributions are determined from the neutron source based on the steady-state balance of the slowing-down and thermal-capture reaction rates.

SIMULATE-E uses the FIBWR steady-state thermal-hydraulic model to predict the flow distribution for a given power distribution. Calculations from the pressure drop analysis provide active and bypass flow distributions. The thermal-hydraulic analysis performed at the beginning of each void iteration provides information about the void distribution. The void distribution determines the nodal cross section values, which in turn determine the thermal power distribution. The relative moderator density of the active coolant is calculated from the thermal power distribution using the void quality profile model. Iterative calculations of the neutron flux, thermal power, and moderator density distributions continue until they are consistent.

Input to SIMULATE-E includes the cross section data provided by NORGE-B, a description of the core loading pattern, the power level, the coolant inlet subcooling and flow rate, and the control rod positions.

SIMULATE-E develops the three-dimensional macroscopic cross section data processed by SIMTRAN-E for input to RETRAN-02. SIMULATE-E also gives k-effectives, nodal power distributions, and nodal exposures. SIMULATE-E has been qualified for use in the reload analysis [Reference 5].

2.1.2 Core Neutronics Linkage Codes

The core neutronics linkage codes, SIMTRAN-E and STRODE, translate SIMULATE-E data into a form that can be used by RETRAN-02. SIMTRAN-E was developed by EPRI [Reference 6]. STRODE was developed by the Supply System.

2.1.2.1 SIMTRAN-E

SIMTRAN-E is a data handling and processing code used for transferring data from SIMULATE-E and NORGE-B to RETRAN-02. SIMTRAN-E reads data from the kinetics parameter tables produced by NORGE-B and from two SIMULATE-E cases. The first SIMULATE-E case models the reactor at the operating state that will represent the initial conditions for the RETRAN-02 transient analysis. The second SIMULATE-E case represents the same state, but with the control rods fully inserted and the void and power reactivity feedbacks disabled to allow a calculation of the scram reactivity. (Applications to transients that do not cause a scram do not require the second SIMULATE-E case.)

SIMTRAN-E radially collapses the three-dimensional cross section and kinetics parameter data read from the SIMULATE-E cases to one dimension by using appropriate weighting functions. Depending on the particular cross section involved, the weighting function is either the volume, the product of the volume and the flux, or the product of the volume and the adjoint flux.

SIMTRAN-E then generates several perturbations of the pressure and fuel temperature about the values obtained from the initial SIMULATE-E case. Each of the perturbations is initially three-dimensional and is collapsed to one dimension by using appropriate weighting functions. The resulting changes in each cross section and each kinetics parameter are represented as polynomial functions of the moderator density and the square root of the fuel temperature. These polynomial fits are subsequently used by RETRAN-02.

2.1.2.2 STRODE

STRODE takes SIMULATE-E data processed through SIMTRAN-E and adjusts the polynomial fits before the data are input to RETRAN-02. This adjustment eliminates differences in core average feedback that otherwise result because SIMULATE-E and RETRAN-02 calculate moderator densities differently.

STRODE uses the results from parallel SIMULATE-E and RETRAN-02 cases to quantify the differences between axial moderator density distributions predicted by the two codes given identical variations in core pressure. STRODE uses the differences between the axial arrays to modify the polynomial coefficients associated with changes in moderator density to obtain consistent moderator density reactivity feedback between SIMULATE-E and RETRAN-02. The STRODE output data is in the same form as that of SIMTRAN-E and is used directly as input to RETRAN-02.

STRODE also corrects the delayed neutron fractions in the one-dimensional data, which are based on the cross section libraries used by CASMO-2E. Delayed neutron fractions in the CASMO-2E cross section library (obtained from ENDF/B-III) are lower than those in the more recent cross section library (ENDF/B-V). The STRODE adjustment makes the one-dimensional data consistent with the more recent data.

2.1.3 System and Core Thermal-Hydraulics Codes

RETRAN-02, the system thermal-hydraulic code, and VIPRE-01, the core thermal-hydraulic code, evaluate core-wide transients for the reload analysis. Both codes were developed by EPRI [References 7,8].

2.1.3.1 RETRAN-02

RETRAN-02 is a neutronic and thermal-hydraulics transient analysis code for modelling nuclear power plants under both normal and

accident conditions. This code can be used either with a point- or with a one-dimensional kinetics representation of the core.

In the reload analysis, RETRAN-02 analyzes dynamic plant transients using one-dimensional neutron kinetics and the core and system-wide thermal-hydraulic models. The RETRAN-02 code solves one-dimensional, two-group, diffusion theory neutron kinetics equations using the space-time factorization method. SIMTRAN-E and STRODE provide the neutronics input to RETRAN-02.

RETRAN-02 is a variable nodalization code requiring user input to specify the system model, which consists of control volumes, heat slabs, and a flow path network. The RETRAN-02 models in this analysis were developed by the Supply System. They were qualified by comparing model predictions with experimental data. Data sources included plant startup tests and the Peach Bottom-2 turbine trip tests. The RETRAN-02 model has been qualified for use in the reload and plant safety analysis [Reference 9].

RETRAN-02 solves conservation of mass, momentum, and energy equations for the fluid mixture. Relative velocities of liquid and vapor are calculated with an algebraic slip model that uses a drift flux approach. Channel friction is calculated using a two-phase friction multiplier to the single-phase friction factor. Heat transfer is calculated with a time-dependent, one-dimensional heat conduction equation. Fill junctions or time-dependent volumes specify the boundary conditions for system thermal-hydraulic calculations.

RETRAN-02 component models represent valves, pumps, and steam separators. Valve opening and closing characteristics are simulated using a table of valve-flow-area versus time. Pump characteristics are simulated with a set of pump curves. Steam separators are modelled with the bubble rise model. Control systems and component trips can also be modelled with RETRAN-02.

2.1.3.2 VIPRE-01

VIPRE-01 analyzes steady-state and transient reactor core thermal-hydraulic conditions. This code also evaluates the MCPR, fuel and cladding temperatures, and coolant conditions. VIPRE-01 can be used for the steady-state or transient analyses of BWR cores with geometries ranging from a single fuel assembly or flow channel up to an entire core. Possible simulations include transient variation of inlet flow, inlet enthalpy, system pressure, average power, local pin power, and power shape.

The transients VIPRE-01 evaluates can range from those found under normal operating conditions to those that develop during moderately severe accidents. Transients occurring entirely within the core can be simulated by VIPRE-01 alone. Transients involving primary system effects outside the core must first, however, be simulated with a system code, such as RETRAN-02, that can provide core boundary conditions to VIPRE-01.

VIPRE-01 predicts three-dimensional velocity, pressure, thermal-energy fields, and fuel rod temperatures for single-phase and two-phase flow. This code also solves the finite difference equations for mass, momentum, and energy conservation for an interconnected array of channels assuming incompressible, thermally expandable, homogeneous flow. The equations are solved with an implicit numerical method that places no restriction on the time step or node size. Although the formulation is homogeneous, it includes models for subcooled boiling and vapor/liquid slip in two-phase flow.

The Supply System uses VIPRE-01 to predict changes in the critical power ratio (ΔCPR) during transients; these changes establish the MCPR operating limit.

2.1.4 Other Codes

The reload analysis also uses a number of supplementary codes to perform specific functions. These codes include ESCORE, FICE, RODDK, TLIM, CALTIP, RBLOCK, and STARS. ESCORE, RODDK, and STARS were developed by EPRI [Reference 10, 11, 12]. RBLOCK was developed by the Yankee Atomic Electric Company [Reference 13]. The remaining codes were developed by the Supply System.

2.1.4.1 ESCORE

ESCORE predicts the steady-state, thermal-mechanical performance of fuel rods. In the reload analysis, ESCORE provides the fuel and clad temperature distributions used in the lattice physics calculations and establishes the gap conductance used in the system and core thermal-hydraulic analyses.

Escore models the fuel rod as a series of discrete axial segments; independent radial thermal equilibrium calculations are performed for each one. The results for each axial segment are coupled through the assumption of the complete mixing of the fill gas and fission gases within the free volume of the fuel rod.

Models within ESCORE describe the fuel pellet and cladding behavior as it is influenced by irradiation history. Fuel pellet models include a steady-state, radial-temperature predictor with flux or power depression, thermal expansion, relocation, densification, swelling, fission gas production and release, and elastic and non-elastic deformation. Cladding models include a steady-state temperature predictor with power deposition, thermal and elastic expansion, creep, and growth.

The fuel pellet is assumed to be a right circular cylinder that responds to thermal- and fission-induced volumetric changes. The fuel cladding conforms elastically and plastically to the calculated conditions of pellet-cladding contact. The internal pressure of the fuel rod is calculated from the amount of fill gas,

the amount of fission gas released, and the free volume of the fuel rod. The fuel temperature depends on the fuel-pellet-to-cladding gap conductance calculated as a function of the gap-gas conductivity and open-gap size or contact load.

2.1.4.2 FICE

FICE is a data handling and processing code used to transfer data from CASMO-2E to the thermal limits code, TLIM, and the traversing incore probe (TIP) calculational code, CALTIP. FICE provides the following data to TLIM:

- o the local peaking functions needed for the critical power ratio (CPR) correlation (the reload analysis currently uses the ANFB correlation)
- o the maximum local peaking factors needed for calculation of linear heat generation rates (LHGR).

Local peaking functions for the ANFB correlation are based on Advanced Nuclear Fuels (ANF) methodology. They are used to account for the local peaking factor, the bundle geometry, and the spacer effects. FICE provides to CALTIP data tables that can be used for interpolating detector signal rate responses.

2.1.4.3 RODDK-E

The RODDK-E code provides rapid estimation of relative control rod worths. RODDK-E uses a two-dimensional, FLARE-based source iteration method to solve the neutron balance, and a FLARE-based source normalization method to reproduce a more detailed SIMULATE-E neutron source distribution. RODDK-E also predicts the relative order of control rod worths, which facilitates the selection of control rods for analysis with the more detailed SIMULATE-E code. SIMULATE-E provides more accurate and more detailed assessments of control rod worth, which reduces some of the uncertainty in the results obtained from RODDK-E.

2.1.4.4 TLIM

TLIM is used to evaluate the thermal margins associated with the results of a SIMULATE-E case. Thermal margins evaluated by TLIM include the following:

- o CPR
- o LHGR
- o average planar linear heat generation rate (APLHGR).

2.1.4.5 CALTIP

CALTIP calculates expected TIP responses for a given core configuration. The core configuration is first modeled with the SIMULATE-E code; CALTIP reads the resulting nodal fluxes, exposures, and other variables from the SIMULATE-E restart file. CALTIP then uses the nodal variables to obtain nodal detector signal rates by interpolating in tables generated by the FICE code from CASMO-2E results. The nodal detector signal rates are then combined and normalized to produce the predicted signal rates for each TIP measurement. CALTIP reads the measured TIP data, compares each measurement with the corresponding predicted value, and prints a statistical summary of the results.

2.1.4.6 RBLOCK

RBLOCK predicts the response of the rod block monitor (RBM) system for specified failure combinations of local power range monitors (LPRM) or LPRM strings. It also identifies the most limiting notch position where a rod block would be predicted to occur for a given limiting failure combination. The RBLOCK code uses individual LPRM responses calculated by the CALTIP code. It also uses TLIM calculations for the MCPR and maximum linear heat generation rate (MLHGR) as a function of notch position. RBLOCK generates an output summary to assist in evaluating compliance with the event acceptance limits for control rod withdrawal errors.

2.1.4.7 STARS

STARS applies the SCU methodology to determine the overall probability distribution for an analysis result in a specific event. The variable of interest is an uncertain quantity with statistical properties estimated by STARS from a response surface/Monte Carlo analysis. The response surface provides a fit to the results of a systems simulation code or code package, such as RETRAN-02/VIPRE-01.

STARS can develop a second-order response surface, with all cross terms, for up to nine parameters. It can also be used to establish selected cubic and quartic dependencies. The Supply System safety analysis uses a second order response surface fit with four or fewer parameters.

Evaluation of a response surface is based on a central composite experimental design technique that identifies a reasonable number of cases to be run with the systems simulation code. Additional data points may be added later to reduce the fitting error. STARS can use up to 350 data points to evaluate the response surface polynomial fitting coefficients, which are determined by a least squares analysis.

2.2 STATISTICAL COMBINATION OF UNCERTAINTIES METHODOLOGY

The safety analysis must consider uncertainties in models, model inputs, instrumentation systems, and the operating state. The approach usually taken assumes that all significant uncertainties are simultaneously at their most adverse points given the possible range of conditions. The plant usually has sufficient margin to accommodate this approach. Occasionally, however, this approach can lead to core operating limits or technical specification setpoints that result in undesirable plant restrictions. In these cases, the safety analysis uses the SCU methodology to define a set of more operationally acceptable setpoints while retaining a sufficient level of conservatism.

The SCU methodology used by the Supply System is based on the approach in the EPRI setpoint analysis guidelines [Reference 14]. This approach consists of the following:

- o identifying the event acceptance criteria and limits that will be used to evaluate the analysis results
- o developing a response surface and applying the methodology.

2.2.1 Event Acceptance Criteria and Limits

The SCU methodology requires determining which part of a system will receive the greatest challenge from a specific event. The SCU analyses must then indicate a 95% probability that the consequences of such an event will be less severe than predicted. Furthermore, the methods of arriving at the probability figure must be sufficiently conservative to ensure a 95% degree of confidence that the figure is correct. In the reload analysis, this degree of confidence is obtained by using deterministic inputs for all parameters except those treated statistically in the development of the response surface or the model uncertainty.
The SCU analysis of anticipated operational occurrences may be used to establish two core operating limits: the operating limit MCPR and the operating limit LHGR. The operating limit MCPR ensures a greater than 95% probability that the MCPR for the fuel cladding integrity (greater than 99.9% of the fuel rods in the core will not be expected to experience boiling transition) will not fall below the specified limit. The operating limit LHGR ensures a greater than 95% probability that PAFF limits (less than 1% plastic strain of the cladding and fuel centerline melt temperature) will not be exceeded. In the safety analysis, the SCU methodology is applied to the operating limit MCPR. The PAFF limits are obtained from the fuel vendor's analysis.

2.2.2 Response Surface Methodology

In the SCU approach, direct stochastic analysis of an event, using event analysis methodology to demonstrate compliance with event

acceptance limits, requires an unacceptable expenditure of resources. Response surface techniques reduce resource requirements to an acceptable level.

Development of a response surface approximating key results of event analysis codes requires five steps:

- o quantifying the model uncertainty
- o selecting and quantifying the parameters for the response surface
- o developing the response surface
- o simulating an event using the response surface
- o convolving the response surface for the event with the model uncertainty.

Quantifying the model uncertainty requires establishing the effect of modelling and input uncertainties on key results of the event analysis. Analysis models are run with perturbed inputs to quantify the effect of specific changes on output. The results are statistically combined to characterize overall model uncertainty. These results are later combined with the results of the response surface analysis to provide an overall probabilistic statement.

Selecting and quantifying parameters requires screening to determine their relative impact on the analysis. The parameters that provide the greatest relative improvement in the evaluation are used for the response surface. Typically, no more than four parameters are used. The Supply System will attempt to limit the number to one or two.

Developing a response surface requires defining the specific system and core thermal-hydraulic cases that must be run to obtain the desired degree of accuracy. A response surface is a polynomial fit to a set of experimental design cases; this fit approximates the results of the system and core thermal-hydraulic analysis model for a specific output parameter over a limited range of interest. The polynomial fitting coefficients that define the response surface

are determined by a least squares fitting technique. As part of the fitting process, the fitting error, which represents the uncertainty introduced into the analysis by the use of the response surface, is determined.

Once a response surface has been defined, the event can be simulated using Monte Carlo techniques. These techniques select a random sample of each independent parameter based on its statistical characteristics. The dependent variable is then calculated from the response surface. Each set of random samples forms one history for statistical evaluation of the dependent variable. Sufficient histories (typically 100,000) will accurately quantify the probability of a given result for a dependent variable.

Convolving the response surface with the model uncertainty establishes an overall probability distribution for the analysis results. This probability distribution is then used to demonstrate compliance with the event acceptance limits.

The computer codes used in the Supply System reload analysis make three contributions to the SCU evaluation:

- perturbation analyses that establish the model uncertainties
- sensitivity studies that identify the parameters to be treated statistically in the development of the response surface
- analyses that determine the shape of the response surface.

A cornerstone in the SCU methodology is the STARS code. This code calculates and evaluates the response surface fitting coefficients based on the experimental design, performs the Monte Carlo assessment of the response surface, and convolves the response surface with the model uncertainty.

Appendix A gives more detail about the application of SCU methodology in the Supply System safety analysis.

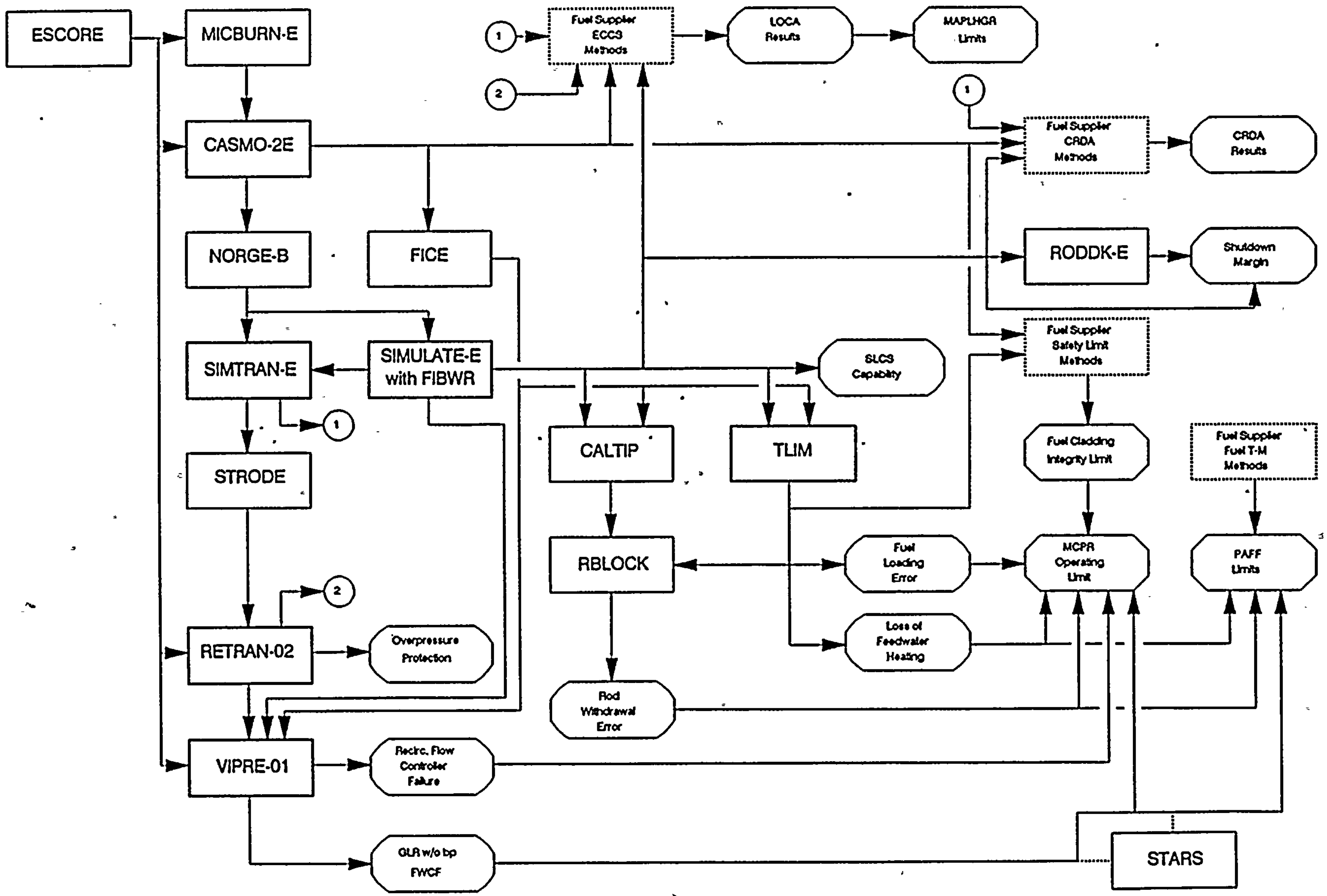

Figure 2-1. Code Sequence for WNP-2 Reactor Analysis Methodology

SECTION 3.0
DESIGN ANALYSIS REQUIREMENTS

This section discusses design criteria, technical bases, supporting analyses, and test results for the thermal-mechanical, nuclear, and thermal-hydraulic design of reload fuel. Fuel design analyses are a cooperative effort between the Supply System and the fuel vendor.

3.1 FUEL THERMAL-MECHANICAL DESIGN

The thermal-mechanical design description includes the following:

- a fuel system description
- the design bases
- a description of mechanical design evaluations
- a summary of supporting test, inspection, and surveillance programs.

The fuel vendor develops the thermal-mechanical design for the reload fuel.

3.1.1 Fuel System Description

The fuel system description includes the following:

- a description of the fuel assembly and its component parts, including fuel rods, water rods, tie plates, and spacers
- fuel assembly drawings with dimensions of all components
- reference design values for all parameters identified in the standard review plan.

The fuel vendor supplies the fuel description as part of the generic licensing of the fuel design. Reference 15 is an example of a fuel vendor report with this information. All fuel designs currently being used by the Supply System for reload fuel have been approved by the NRC.

3.1.2 Design Bases

The design bases provide the criteria for acceptable fuel assembly performance during normal operation, anticipated operational occurrences, and postulated accidents. Design bases are developed for both the fuel rods and the fuel assembly. Fuel rod designs must comply with regulatory requirements for the following:

- rod internal pressure
- cladding strain
- cladding temperature
- fuel temperature
- cladding fretting wear
- cladding fatigue
- cladding corrosion
- hydriding
- cladding collapse
- fuel enthalpy.

Fuel assembly designs must comply with regulatory requirements for the following:

- structural integrity
- corrosion
- hydriding
- dimensional compatibility
- hydraulic loads
- dimensional stability
- shipping and handling loads
- fuel coolability.

3.1.3 Mechanical Design Evaluations

Design evaluations ensure that all bases for a proposed reload fuel design and core configuration have been satisfied. The results of these evaluations are documented in the reload summary report. Licensing topical reports submitted to the NRC describe the models

used in specific design analyses. The results of design evaluations are used to set core operating limits.

3.1.4 Test, Inspection, and Surveillance Programs

Testing, inspection, and surveillance programs ensure the operational acceptability of proposed reload fuel designs. These programs are described in generic topical reports submitted by the fuel vendor to the NRC. Reference 16 is an example of such a report.

3.2 NUCLEAR DESIGN

The nuclear design description includes the following:

- a reload fuel description
- the design bases
- a description of nuclear design evaluations
- a description of the reload fuel design methodology
- the nuclear input parameters used in the safety analysis.

The reload summary report and the licensing topical reports that support it provide the nuclear design bases necessary to approve the proposed reload fuel design.

3.2.1 Reload Fuel Description

The reload fuel description includes the following:

- the enrichment distribution in the assembly
- the amount and location of burnable poison in the assembly
- the moderator distribution within each fuel assembly.

The reload summary report documents this information for each reload.

3.2.2 Design Bases

Nuclear design bases ensure that functional and regulatory requirements for nuclear design are satisfied. Nuclear designs must comply with regulatory requirements for the following:

- o fuel burnup
- o reactivity coefficients
- o control of power distribution
- o shutdown margin
- o criticality criteria for fuel storage
- o stability.

3.2.3 Nuclear Design Evaluations

Design evaluations ensure that all nuclear design bases for the proposed reload fuel and core configuration have been satisfied.

3.2.4 Reload Fuel Design Methodology

The description of the reload core design methodology documents the approach to the reload core design. The description includes a reference core design for the proposed reload and all analyses required for licensing.

3.2.5 Nuclear Input Parameters Used in the Safety Analysis

Nuclear input parameters are an important aspect of a safety analysis. These parameters include

- o core kinetics characteristics
- o scram reactivity
- o core power distributions
- o core neutron cross sections and their dependencies on the fuel temperature and coolant void level.

Each safety analysis model requires a slightly different set of input parameters. These parameters are evaluated for core conditions that make the consequences of a particular event most severe.

3.3 THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic design description includes the following:

- a thermal-hydraulic system description
- the design bases
- a description of thermal-hydraulic design evaluations.

The reload summary report and the licensing topical reports that support it provide the thermal limits necessary to approve the proposed reload fuel design.

3.3.1 Thermal-Hydraulic System Description

The thermal-hydraulic system description includes the following:

- a list of key hydraulic parameters, such as base rod flow area, hydraulic diameter, and heated surface area of the reload fuel
- component loss coefficients that permit hydraulic modelling of the reload fuel in a mixed core with co-resident fuel assemblies.

3.3.2 Design Bases

Thermal-hydraulic design bases ensure that thermal-hydraulic functional and regulatory requirements are satisfied. Thermal-hydraulic designs must meet regulatory requirements for the following:

- thermal and hydraulic compatibility
- bypass flow analysis

- water-rod analysis
- stability.

The thermal design basis is that greater than 99.9% of the fuel rods are not expected to experience boiling transition during normal operation or anticipated operational occurrences. Core operating limits that satisfy this design basis are derived using transient analysis ΔCPR methods and a reactor statistical safety limit methodology [Reference 17].

Hydraulic compatibility between reload fuel assemblies and resident fuel assemblies must be demonstrated whenever reload fuel differs hydraulically from resident fuel. Furthermore, bypass-flow and water-rod analyses must demonstrate that both the bypass region and the water rods will remain sufficiently cool during expected operating conditions. The design bases are that water in the bypass region and in the water rods will remain essentially subcooled during normal operation.

3.3.3 Thermal-Hydraulic Design Evaluations

Design evaluations ensure that all bases for a proposed reload fuel design and core configuration have been satisfied. Reload summary reports document the results of design evaluations and address regions of instability.

The thermal-hydraulic analyses are performed on a cycle-by-cycle basis. Technical specifications implemented at WNP-2 define regions of instability and preclude operation of the reactor in these regions.

SECTION 4.0
SAFETY ANALYSIS REQUIREMENTS

Safety analyses determine the safety margin during normal operation, anticipated operational occurrences, accidents, and special events. They also ensure that any proposed changes to plant configuration satisfy all safety requirements, licensing commitments, and regulatory guidelines and requirements. Furthermore, safety analyses identify any changes in operating limits or technical specifications that will be required as a result of changes in plant configuration.

Previous safety analyses demonstrate that WNP-2 can operate safely in its current configuration. The Supply System methodology will be used to evaluate any changes to that configuration, such as the insertion of reload fuel.

4.1 SAFETY ANALYSIS CATEGORIES

Safety analyses evaluate a wide spectrum of plant events and conditions. Different acceptance criteria and limits are applied to events with different probabilities of occurrence. For convenience, these events and conditions have been divided into categories that reflect the probability of occurrence and analysis requirements:

- normal operation and anticipated operational occurrences
- accidents
- special events.

4.1.1 Normal Operation and Anticipated Operational Occurrences

Normal operation encompasses all modes of planned plant operation, including startup, operation, shutdown, and refueling. Furthermore, all events that require analysis are assumed to initiate from some mode of normal operation. Normal operation therefore provides the initial conditions for the transient safety

analysis, which evaluates the anticipated operational occurrences with parameters that pose the most significant challenge to the fuel or reactor coolant pressure boundary capabilities. In the Supply System methodology, these occurrences fall into eight categories:

- o decrease in reactor coolant temperature
- o increase in reactor pressure
- o decrease in reactor coolant flow rate
- o reactivity and power distribution anomalies
- o increase in reactor coolant inventory
- o decrease in reactor coolant inventory
- o increase in reactor coolant flow
- o increase in reactor coolant temperature.

These occurrences include all those identified in applicable regulatory requirements and guidelines. Each occurrence is discussed below.

4.1.1.1 Decrease in Reactor Coolant Temperature

Events that directly decrease the reactor coolant temperature are those that either increase the flow of cold water or reduce the temperature of water being delivered to the reactor vessel. Reducing the reactor coolant (moderator) temperature increases core reactivity, which in turn increases core power. The resulting negative moderator void reactivity shifts power towards the bottom of the core. These changes will lead to a new steady-state power level, which will require corrective action by the operator. Sufficiently high levels of thermal power or neutron flux will cause a scram. Events in this category include

- o loss of feedwater heating
- o inadvertent high pressure core spray (HPCS) startup
- o inadvertent residual heat removal (RHR) shutdown cooling operation.

4.1.1.2 Increase in Reactor Pressure

Events that increase reactor pressure significantly are usually initiated by a sudden reduction in steam flow. The increased pressure collapses the voids in the core, which increases core reactivity. This causes an increase in the core power level, which further increases core pressure. A scram will terminate this event. Safety analysis events in this category include

- o digital-electric-hydraulic (DEH) pressure regulator failure in the closed position
- o generator load rejection
- o turbine trip
- o closure of the main-steam-line isolation valve.
- o loss of the condenser vacuum.

4.1.1.3 Decrease in Reactor Coolant Flow Rate

Events that reduce recirculation flow also reduce the reactor coolant flow rate, which increases core voids and decreases core reactivity. The decrease in reactor coolant flow increases the water level because of the swelling of moderator voids. The increase in core voids decreases the power level. Under certain conditions, the increase in water level may cause a turbine and feedwater trip, which will, in turn, initiate a reactor scram and require operation of the high pressure makeup systems. Events in this category include

- o recirculation pump trip (RPT)
- o recirculation flow control failure in the decreasing flow position.

4.1.1.4 Reactivity and Power Distribution Anomalies

Localized reactivity increases and anomalies in the power distribution are usually due to operator errors involving control rod movement. The effects of the changes vary. A large reactivity

addition may require a reactor scram or rod block to terminate the event. Lack of corrective action will lead to a new steady-state operating condition. Events in this category include

- o control rod withdrawal error at low power
- o control rod withdrawal error at full power.

4.1.1.5 Increase in Reactor Coolant Inventory

Events that lead to a feedwater flow rate higher than the steam production rate increase the amount of water (coolant inventory) in the reactor vessel, and may initiate a turbine and feedwater trip on high water level. A turbine trip will, in turn, result in increased core pressure, with a concomitant void collapse and reactivity increase. The resulting increase in power level will be terminated by the reactor scram initiated by the turbine trip. A feedwater trip may also lead to conditions that actuate the high pressure makeup systems. The one event in this category is the feedwater controller failure.

4.1.1.6 Decrease in Reactor Coolant Inventory

Events that lead to a steam production rate that is higher than the feedwater flow rate decrease the water level in the reactor vessel. If the feedwater flow is sufficient to maintain the vessel water at a new level, a new steady-state operating condition will be established until operator action is taken. Otherwise, the event will be terminated by a scram on low water level. Events in this category include

- o inadvertent opening of the safety relief valve
- o DEH pressure regulator failure in the open position
- o loss of feedwater flow

4.1.1.7 Increase in Reactor Coolant Flow

Events that increase recirculation flow also increase the reactor coolant flow rate, which decreases coolant temperature and voids. These changes cause an increase in core reactivity and an increase in power level. A slow increase in coolant flow may lead to a new steady-state operating condition, which can be terminated by operator action. A rapid increase will initiate a scram on high neutron flux. Events in this category include

- recirculation flow control failure in the increasing flow position
- startup of an idle recirculation pump.

4.1.1.8 Increase in Reactor Coolant Temperature

The one event in this category is the failure of the RHR shutdown cooling system. This event increases the reactor coolant temperature, which leads to a slow increase in pressure. The shutdown cooling system is assumed to be isolated. Operator action must establish an alternate coolant path through the low pressure coolant injection line.

4.1.2 Accidents

Accidents are postulated events that would affect one or more of the radioactive material barriers. These events are not expected to occur during the lifetime of the plant. They are, however, used to establish design bases for some systems. Accidents fall into eight categories:

- control rod drop accidents
- main steam line breaks
- instrument line pipe breaks
- LOCAs
- fuel handling accidents
- recirculation pump seizures or shaft breaks

- radwaste system failures
- fuel loading errors.

Each of these categories is discussed below.

4.1.2.1 Control Rod Drop Accident

Rapid removal of a high-worth control rod may create a power excursion significant enough to affect the fuel cladding and reactor coolant pressure boundary. The postulated control rod drop accident is the dropping of a fully inserted and decoupled control rod at maximum velocity. The control rod is assumed to be the maximum-worth rod consistent with the constraints on the control rod patterns. This type of accident creates a sudden burst of power, which is initially limited by the Doppler reactivity. A control rod scram initiated on high neutron flux will terminate the event.

4.1.2.2 Main Steam Line Break Accident

Some systems that penetrate the primary and secondary containment are connected to the reactor coolant pressure boundary. The postulated main steam line break is a break in a steam line outside the primary containment, where it releases the maximum amount of reactor coolant directly to the environment. The break causes rapid depressurization of the reactor and an increase in the void fraction, which shuts down the nuclear reaction. The main steam isolation valves will then close and terminate the event.

4.1.2.3 Instrument Line Break

A number of instrument lines that penetrate the primary containment connect directly to the reactor coolant pressure boundary. The postulated instrument line break is the complete severance of an instrument line outside the primary containment and upstream from the isolation valve. A break in this location cannot be isolated and may release a significant amount of reactor coolant to the

environment. The reactor will continue to operate until the break is detected. The reactor must be shut down and the system depressurized to minimize releases until the break is repaired.

4.1.2.4 Loss of Coolant Accident

A number of pipes penetrate the reactor coolant pressure boundary. The analysis of the postulated LOCA includes the evaluation of a spectrum of pipe break sizes and locations. These breaks cause coolant to leak from the reactor and be discharged into the primary containment. Depressurization along with a low-water level will initiate a reactor scram, close the containment isolation valves, and activate the emergency core cooling system (ECCS) and other required equipment.

4.1.2.5 Fueling Handling Accident

Accidents can release radioactivity directly to the secondary containment when the primary containment is not intact. The greatest potential for a release of this type occurs during refueling, when the head is off the reactor vessel and the primary containment is open.

The postulated fuel handling accident is the dropping of a fuel assembly onto the core or onto fuel in the fuel storage pool in a way that maximizes damage to the exposed fuel. This type of accident will activate the secondary containment isolation system and the standby gas treatment system.

4.1.2.6 Recirculation Pump Seizure or Shaft Break

Failure of a recirculation pump rapidly reduces the core coolant flow. The postulated recirculation pump failure is that of a pump seizing or a shaft breaking. The level swell caused by the rapid decrease of coolant flow trips the main and feedwater turbines and produces a stop valve closure scram and an RPT. The bypass and safety/relief valves will keep the vessel pressure within the

American Society of Mechanical Engineers (ASME) Code limits [Reference 18].

4.1.2.7 Radwaste System Failures

The postulated radwaste system failures lead to the maximum amount of radioactivity being released directly to the environment. The plant will continue to operate until the failure is detected. Operator action will be required to isolate the system and minimize releases.

4.1.2.8 Fuel Loading Error

The postulated fuel assembly loading error is that of placing one fuel assembly in the wrong location or rotating it. For this analysis, the following assumptions have been made:

- Only one fuel assembly is incorrectly loaded.
- The loading error is not detected before startup through the core verification process.
- The loading error is not detected after startup by the process instrumentation.
- The plant is capable of operating at thermal limits throughout the operating cycle.

The fuel loading error is a highly localized event that significantly affects only the misloaded assembly and adjacent assemblies. Although fuel loading errors are classed as accidents, the Supply System applies to this type of error the stringent operational criterion of ensuring that fuel cladding integrity limits are met.

4.1.3 Special Events

Special event analyses ensure compliance with specific regulatory and licensing requirements not considered in the analyses of normal operation, anticipated operational occurrences, and accidents. Six

categories typically evaluated in a BWR reload analysis include the following:

- o shutdown margin
- o standby liquid control system capability
- o Code overpressure protection analysis
- o stability
- o shutdown from outside the control room
- o anticipated transients without scram.

The special events are described below.

4.1.3.1 Shutdown Margin

The shutdown margin analysis demonstrates that the core can be made subcritical with sufficient margin when the highest-worth control rod is in the full-out position and the remaining control rods are fully inserted. For this analysis, the core is assumed to be at cold conditions, with no xenon present, and at the most reactive stage in the operating cycle.

4.1.3.2 Standby Liquid Control System Capability

The standby liquid control system capability analysis demonstrates that the core can be made subcritical by actuating the standby liquid control system. The analysis verifies that the core becomes subcritical at all power conditions with minimum control rod inventory and no xenon present.

4.1.3.3 Code Overpressure Protection Analysis

The Code overpressure protection analysis confirms conformance with the ASME Code requirement for protecting against overpressure [Reference 18]. The Code overpressure protection analysis usually simulates the most severe pressurization event in an anticipated operational occurrence with an indirect scram that is not associated with the event initiator.

4.1.3.4 Stability

Stability evaluations ensure that no divergent power oscillations will occur that cannot be detected and suppressed. The safety analysis evaluates three types of hydrodynamic stability: plant, core, channel.

4.1.3.5 Shutdown from Outside the Control Room

This evaluation ensures that the reactor can be shut down from outside the control room. It employs an operator procedure for reaching cold shutdown state using only equipment outside the control room.

4.1.3.6 Anticipated Transients Without Scram

The postulated anticipated transient without scram is an anticipated transient that reaches a reactor protection system (scram) setpoint or requires a manual scram without enough insertable control rods to shut the reactor down. Transients of this nature are expected to occur one or more times during the life of the plant. The Supply System uses an RPT, a standby liquid control system, and alternate rod insertion (ARI) to protect WNP-2 against failure to scram.

4.2 SAFETY ANALYSIS CRITERIA AND LIMITS

As noted in section 4.1, three categories of events require a safety analysis. The Code of Federal Regulations (CFR) specifies the requirements for analyzing these events. The General Design Criteria (GDC), Appendix A, 10CFR50, list many of the requirements. These requirements vary widely because of the differences in the nature of different events and the difference in the probabilities of occurrence. Table 4-1 lists the boundaries, criteria, and limits for the three types of events. Table 4-2 lists specific events and limits. The discussion below summarizes the federal requirements.

4.2.1 Normal Operation & Anticipated Operational Occurrences

Normal operation and anticipated operational occurrences require criteria and limits for the following:

- site
- fuel cladding integrity
- reactor coolant pressure boundary integrity

Site requirements are less limiting than those for fuel cladding integrity and reactor coolant pressure boundary integrity. Therefore, it is unnecessary to deal with site requirements as long as requirements for the last two categories are satisfied.

4.2.1.1 Fuel Cladding Integrity

The GDC-10 governs fuel cladding integrity limits. It requires that the reactor core and associated coolant, control, and protection systems be designed with a margin sufficient to ensure that specified acceptable fuel design limits (SAFDL) are not exceeded during normal operation or anticipated operational occurrences. The SAFDLs place limits on cladding stress or strain, fuel temperature, fuel enthalpy, and cladding overheating.

The SAFDL for cladding stress or strain comes from the ASME Code, which specifies that fuel cladding shall experience no more than 1% plastic strain [Reference 18].

The SAFDL for fuel temperature is the maximum LHGR that will not lead to fuel centerline melting. This limit is a function of exposure.

The SAFDL for fuel enthalpy is less than 170 cal/gm radially averaged at any axial location for reactivity insertion events initiated from low power. In the Supply System reload analysis, this SAFDL is used only to estimate the number of fuel rods assumed to fail in the control rod drop accident analysis.

The SAFDL associated with cladding overheating is that greater than 99.9% of the fuel rods are not expected to experience boiling transition.

4.2.1.2 Reactor Coolant Pressure Boundary Integrity

The GDC-15 requires that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to ensure that design limits for the reactor coolant pressure boundary will not be exceeded during normal operation or anticipated operational occurrences. The ASME Code specifies a peak pressure of less than 1375 psig within the reactor coolant pressure boundary [Reference 18].

4.2.2 Accidents

Postulated accidents require criteria and limits for the following:

- site
- reactor coolant pressure boundary integrity
- primary containment integrity.

Site requirements are less limiting than those for reactor coolant pressure boundary integrity and primary containment integrity. Therefore, it is unnecessary to deal with site requirements as long as requirements for the last two categories are satisfied.

4.2.2.1 Reactor Coolant Pressure Boundary Integrity

The GDC-14 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested to ensure an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The ASME Code specifies limits [Reference 18].

The GDC-28 requires that reactivity control systems be designed to limit the amount and rate of reactivity increase to ensure that postulated reactivity accidents will cause no damage to the reactor

coolant pressure boundary greater than limited local yielding. Current regulatory guidelines limit the calculated peak radial average enthalpy to 280 cal/gm at any axial location in the fuel and the peak reactor vessel pressure to less than the emergency stress limits allowed by the ASME Code [Reference 18].

4.2.2.2 Primary Containment Integrity

The 10CFR50.44, GDC-16, and GDC-50 specify containment design criteria. The ECCS capability criteria and performance limits ensure that other containment integrity limits will be satisfied by ensuring that postulated containment loads and core damage are bounded by previous assumptions in the plant safety analysis.

The GDC-35 specifies that the ECCS must be capable of transferring heat from the reactor core following any LOCA at a rate sufficient to ensure negligible fuel and fuel cladding damage that could interfere with effective core cooling. The 10CFR50.46 specifies the following ECCS limits:

- a peak cladding temperature ≤ 2200°F
- a maximum cladding oxidation ≤ 17%
- core-wide metal-water reaction ≤1%
- the maintenance of a coolable geometry
- assurance of long term cooling.

4.2.3 Special Events

Special events require criteria and limits for the following:

- fuel cladding integrity
- reactor coolant pressure boundary integrity

4.2.3.1 Fuel Cladding Integrity

The GDC-12 requires that the reactor core and associated coolant, control, and protection systems be designed to ensure that power oscillations that could damage fuel are not possible or can be reliably and readily detected and suppressed. The GDC-19 requires that equipment be placed at appropriate locations outside the control room with a design capability for achieving hot shutdown with a potential for cold shutdown.

The GDC-26 requires two independent reactivity control systems of different design principles. One of the systems must use control rods and must ensure that the SAFDLs will not be exceeded during normal operation or anticipated operational occurrences. This system must also allow an appropriate margin for malfunctions, such as stuck control rods. The second system must ensure that the SAFDLs will not be exceeded during planned, normal power changes, including xenon burnout.

The GDC-27 requires that the reactivity control systems have a combined capability, with appropriate margin for stuck control rods, sufficient to control reactivity changes under postulated accident conditions. One of the systems must also be capable of holding the reactor core subcritical under cold conditions.

4.2.3.2 Reactor Coolant Pressure Boundary Integrity

The 10CFR50.55a specifies that ASME Code pressure limits [Reference 18] apply to Code overpressure protection analysis. The 10CFR50.62 identifies design features required for anticipated transients without scram. This regulation also mandates that sufficient information be provided to the NRC to demonstrate the suitability of these design features.

Table 4-1

SAFETY ANALYSIS BOUNDARIES, CRITERIA, AND LIMITS

NORMAL OPERATION AND ANTICIPATED OPERATIONAL OCCURRENCES

BOUNDARY	CRITERIA	LIMITS
Site	Limits on Release to Unrestricted Areas (10CFR20)	Numerical Limits for Release of Radioactive Materials (10CFR20)
	Releases as Low as Reasonably Achievable (10CFR50.34a)	Numerical Guides and Dose Objectives (10CFR50, Appendix I)
Fuel Cladding	SAFDLs (GDC-10)	Fuel Cladding Integrity Limit (SRP 4.4)
		Fuel Cladding Plastic Strain Design Limit (SRP 4.2)
		Peak Fuel Enthalpy Limit (SRP 4.2)
		Fuel Centerline Melt Limit (SRP 4.2)
Reactor Coolant Pressure Boundary	Reactor Coolant Pressure Boundary Design Limits (GDC-15)	Nuclear System Design Limits for Upsets (10CFR50.55a)

Table 4-1 (continued)

SAFETY ANALYSIS BOUNDARIES, CRITERIA, AND LIMITS

ACCIDENTS

BOUNDARY	CRITERIA	LIMITS
Site	Site Limits (10CFR100.10)	Guideline Dose Values (10CFR100.11)
	Operator Exposure Limits (GDC-19)	Exposure Limits for Plant Operators (GDC-19)
Reactor Coolant Pressure Boundary	Reactor Coolant Pressure Boundary Design (GDC-14)	Containment Design Limits for Accidents (10CFR50.55a)
	Reactivity Insertion Limits (GDC-28)	Peak Fuel Enthalpy Limit (Reg. Guide 1.77)
Primary Containment Boundary	Containment Design (GDC-16&50 & 10CFR50.44)	Nuclear System Design for Accidents (10CFR50.55a)
	Emergency Core Cooling System Capability (GDC-35)	Emergency Core Cooling System Performance Limits (10CFR50.46)

Table 4-1 (continued)

SAFETY ANALYSIS BOUNDARIES, CRITERIA, AND LIMITS

SPECIAL EVENTS

BOUNDARY	CRITERIA	LIMITS
Fuel Cladding	Reactivity Control System Design (GDC-26&27)	Shutdown Margin Limit (SRP 4.3)
	Shutdown Capability (GDC-19)	Cold Reactor Shutdown from Outside Control Room (GDC-19)
	Reactor Power Oscillation Control (GDC-12)	Suppression of Reactor Power Oscillations (GDC-12)
Reactor Coolant Pressure Boundary	Reactor Coolant System Design (GDC-15)	ASME Code Limits (10CFR50.55a)
	Anticipated Transients Without Scram Criteria (10CFR50.62)	Limits Associated With Plant Systems Performance (10CFR50.62)

Table 4-2

SAFETY ANALYSIS EVENTS AND LIMITS

ANTICIPATED OPERATIONAL OCCURRENCES

EVENT	LIMIT
Loss of Feedwater Heating	SAFDL
Inadvertent High Pressure Core Spray Startup	SAFDL
Inadvertent RHR Shutdown Cooling Operation	SAFDL
Pressure Regulator Failure-Closed	SAFDL/Pressure
Generator Load Rejection	SAFDL/Pressure
Turbine Trip	SAFDL/Pressure
Main Steam Line Isolation Valve Closure	SAFDL/Pressure
Loss of Condenser Vacuum	SAFDL/Pressure
Recirculation Pump Trip	SAFDL
Recirculation Flow Controller Failure-Decreasing Flow	SAFDL
Control Rod Withdrawal Error-Low Power	SAFDL
Control Rod Withdrawal Error at Power	SAFDL
Feedwater Controller Failure	SAFDL/Pressure
Inadvertent Safety Relief Valve Opening	SAFDL
Pressure Regulator Failure-Open	SAFDL
Loss of Feedwater Flow	SAFDL
Loss of ac Power	SAFDL/Pressure
Recirculation Flow Controller Failure-Increasing Flow	SAFDL
Startup of Idle Recirculation Pump	SAFDL
Failure of RHR Shutdown Cooling	Not Applicable

Note:
The SAFDLs require
o less than 1% fuel rod cladding strain
o no fuel centerline melt
o peak fuel enthalpy less than 170 cal/gm
o greater than 99.9% of the fuel rods not expected to experience boiling transition.
Complying with the SAFDLs makes it unnecessary to address the site limits for normal operation and anticipated operational occurrences.

Table 4-2 (continued)

SAFETY ANALYSIS EVENTS AND LIMITS

ACCIDENTS

EVENT	LIMIT
Control Rod Drop Accident	Offsite and Onsite Doses Peak Fuel Enthalpy Reactor Vessel Pressure
Main Steam Line Break	Offsite and Onsite Doses
Instrument Line Break	Offsite and Onsite Doses
Fuel Handling Accident	Offsite and Onsite Doses
Loss of Coolant Accidents	Offsite and Onsite Doses ECCS Acceptance Criteria Containment Design Limits
Recirculation Pump Seizure or Shaft Break	Offsite and Onsite Doses
Radwaste System Failures	Offsite and Onsite Doses
Fuel Loading Error	Fuel Cladding Integrity

Table 4-2 (continued)

SAFETY ANALYSIS EVENTS AND LIMITS

SPECIAL EVENTS

EVENT	LIMIT
Shutdown Margin	Shutdown Margin
Standby Liquid Control System Capability	Cold Reactor Shutdown
Overpressure Protection Analysis	Pressure
Stability	*
Shutdown from Outside Control Room	Reactor Shutdown from Outside the Control Room
Anticipated Transients without Scram	10CFR50.62

* Code of Federal Regulations, 10CFR50, Appendix A (GDC-12) requires that the reactor core and the associated coolant control and protection systems be designed to ensure that power oscillations, which can result in conditions exceeding SAFDL limits, are not possible or can be reliably and readily detected and suppressed.

SECTION 5.0
RELOAD ANALYSIS

The reload analysis uses approved fuel designs from the fuel supplier and, for some reload fuel analyses, integrates fuel supplier analysis methods into the Supply System reload analysis methodology. This section discusses the following:

- the reload analysis process
- the relationship between the reload analysis and the plant safety analysis requirements
- the reload evaluation requirements
- the safety limits used in the analysis process
- the process used to identify events that are not limiting for purposes of the reload analysis.

5.1 INTRODUCTION

A reload is defined as replacement of depleted or exposed fuel assemblies with sufficient fresh fuel assemblies or other high reactivity (reinsert) fuel assemblies to enable another cycle of operation to proceed. The reload analysis demonstrates that plant safety analysis requirements have been satisfied for the reload fuel assemblies in the new core configuration.

5.1.1 Reload Analysis Process

Figure 5.1-1 provides an overview of the Supply System reload analysis.

The first task in the reload analysis is to select and procure the number and type of fuel assemblies needed for the reload. The choice is based on the characteristics of the current core and fuel designs and on an energy utilization plan. The energy utilization plan indicates the expected operating cycle length, the target plant capacity factor, and the anticipated end-of-cycle exposure for the current operating cycle. Fuel cycle studies are then

performed, and a reference fuel cycle is developed that specifies the number and type of reload fuel assemblies necessary to economically satisfy the energy utilization plan. Once the Supply System has chosen the number of fuel assemblies, the fuel design, and the fuel enrichment, fuel fabrication can begin.

The next step in the reload analysis process is to establish a reference loading pattern that will serve as the basis for the reload fuel design and safety analyses. The reference loading pattern identifies the type and location of all fuel assemblies that will be used in the reactor core during the next operating cycle. Important characteristics that influence the reference loading pattern are the predicted end-of-current-cycle core exposure, exposure distribution, and operating history. Section 7.0 discusses parameters that must be checked if the core characteristics at the end of the cycle differ from those predicted and/or if the actual loading pattern differs from the reference loading pattern.

The next step in the reload analysis is to identify the potentially limiting events in the plant safety analysis. This process requires identifying and listing current safety analysis events applicable to the reload analysis. The relationship between the safety analysis and the reload analysis is described in more detail in section 5.1.2. The reload evaluation requirements are described in section 5.1.3. The reload analysis limits are established using fuel vendor design analyses and current plant safety analysis requirements. Section 5.1.4 describes the methodology used to obtain the limits. The reload analysis limits make it possible to identify potentially limiting events by eliminating non-limiting events from the list of applicable events. Section 5.1.5 describes the process for identifying non-limiting events.

The potentially limiting events are the subject of the event analyses. Event analysis requirements are described in more detail in section 5.2. Specific event analysis procedures and their bases are described in section 5.3.

The results of the reload analysis are documented in the reload summary report. (Section 6.0 provides an outline of a reload summary report.) The reload summary report is used to update the plant safety analysis, define the required core operating limits, and provide the basis for any license amendment requests.

The core operating limits are based on the results of the event analyses. Section 6.0 describes how the core operating limits are identified.

In some cases, the results of the analysis may lead to a change in plant technical specifications. Any plant technical specification change requires a license amendment request. Section 6.0 discusses considerations associated with technical specification changes.

The final step in the reload analysis is confirming the suitability of the actual loading pattern. Section 7.0 gives more details about the confirmation process.

5.1.2 Relationship to Plant Safety Analysis Requirements

Section 4.0 described the safety analysis requirements with respect to event analyses. The introduction of reload fuel into the core can change the results of some of the event analyses performed for previous cycles. Any such changes must be evaluated.

Not all events included in the plant safety analysis are significantly impacted by the introduction of reload fuel. Some events in each category may, however, require reanalysis if the changes to the nuclear and thermal hydraulic characteristics of the core are significant.

A screening process identifies and eliminates the non-applicable and non-limiting events. Section 5.1.5 describes the process of identifying potentially limiting events. Section 5.2 describes the application and results of the screening process.

The remaining events are the subject of analysis for the reload fuel and core design. Reload fuel evaluation requirements are described in section 5.1.3.

5.1.3 Reload Evaluation Requirements

The reload analysis updates the plant safety analysis. To ensure a complete assessment, each event in the plant safety analysis must be evaluated to determine the extent to which it challenges the safety limits.

Potentially limiting anticipated operational occurrences are generally analyzed to establish the core operating limits for the reload fuel and core configuration. Accidents may be reanalyzed to establish operating limits for reload fuel or to assess the event consequences if the bounding conditions are not satisfied. Special events are analyzed to demonstrate conformance to specific regulatory requirements. Considerations used to determine the need for and extent of event analyses are described in section 5.2.

Event limits are a key consideration in determining which events require reanalysis for reloads. Section 5.1.4 describes the specific event limits and the process used to establish them.

5.1.4 Safety Limits Used in the Analysis Process

Section 4.0 describes criteria and limits applied to event analysis These criteria and limits are as applicable to the reload analysis as they are to the current plant safety analysis. They serve two purposes:

- o They provide the figures of merit for the reload analysis results.
- o They are used in event screening to assess the relative severity of the events in each category.

Applicable limits for the events in each category are provided in Table 5.1-1; the bases are discussed in sections 5.1.4.1 through 5.1.4.3. It is only necessary to check event analyses results against the most restrictive limits for the event.

5.1.4.1 Anticipated Operational Occurrences

Three sets of limits apply to anticipated operational occurrences:

- o onsite and offsite dose limits associated with the release of radioactive materials
- o design limits for the reactor coolant pressure boundary
- o SAFDLs.

The onsite and offsite dose limits are not used in the reload analysis because the SAFDLs are more limiting. Onsite and offsite dose limits are based on the assumption of a limited number of fuel failures in the core, consistent with plant technical specification requirements. No significant number of fuel failures will occur as long as the SAFDLs are not exceeded. Therefore, previous dose assessments remain applicable, and no further evaluations are required for the reload analysis.

The reactor coolant pressure boundary design limits are based on the requirements in the ASME Code [Reference 18]. These requirements specify a peak vessel pressure limit of 1375 psig.

Four limits apply to SAFDLs:

- o the fuel cladding integrity limit
- o the fuel cladding plastic strain design limit
- o the fuel centerline melt limit
- o the peak fuel enthalpy design limit.

The process of establishing these limits is described below.

5.1.4.1.1 Fuel Cladding Integrity Limit

The fuel cladding integrity limit ensures that greater than 99.9% of the fuel rods in the core would not be expected to experience boiling transition. The Supply System reload analysis will use the fuel vendor methodology and approved critical power or critical heat flux correlations to establish a MCPR that satisfies this requirement.

Currently, the only CPR correlation used with the fuel vendor methods is the approved ANFB correlation [Reference 19]. Analyses using ANFB include the effects of channel bow in the MCPR limit.

5.1.4.1.2 Fuel Plastic Strain Design/Fuel Centerline Melt Limits

The fuel plastic strain limit is the LHGR that produces a permanent fuel cladding deflection of 1%. The fuel centerline melt limit is the LHGR required to reach the fuel melting temperature at the centerline of a fuel pellet. The Supply System methodology uses the fuel vendor's thermal-mechanical methods to establish the LHGR limits for fuel plastic strain and fuel centerline melt limits. These limits are expressed as a function of exposure. To evaluate current fuel designs, these two limits have been combined into a single set of PAFF limits that use the more limiting LHGR.

5.1.4.1.3 Peak Fuel Enthalpy Design Limit

The safety analysis identifies a peak fuel enthalpy design limit for energy deposition from reactivity insertion events initiated from low power. In the Supply System methodology, this limit is an axially averaged peak fuel enthalpy of 170 cal/gm. It is used only to determine the number of fuel rods predicted to fail as a result of a control rod drop accident. This limit does not apply to other events analyzed in the reload analysis process.

5.1.4.2 Accidents

In the plant safety analysis, four sets of limits apply to accidents:

- o offsite guideline dose values and operator exposure limits
- o reactor coolant pressure boundary integrity limits
- o primary containment integrity limits
- o for the fuel loading error, the fuel cladding integrity limit.

The offsite guideline dose values and operator exposure limits are not used in the reload analysis because they are less restrictive than the limits established from bounding analyses for accidents. Section 5.2 discusses the basis for bounding in the event analyses.

Two sets of limits apply to the reactor coolant pressure boundary:

- o nuclear system design limits for accidents, which are based on ASME Code requirements for emergency or faulted conditions
- o peak fuel enthalpy limits for reactivity insertion accidents.

The nuclear system design limits for accidents are not used because bounding analyses exist in the current plant safety analysis. Section 5.2 discusses the basis for the current safety analysis being bounding. A peak fuel enthalpy of 280 cal/gm is the limit for reactivity insertion accidents. The results of analyses for the limiting reactivity insertion event demonstrate that the ASME Code limits for the reactor coolant pressure boundary will not be exceeded for a peak fuel enthalpy of 280 cal/gm.

Two sets of limits apply to the primary containment integrity:

- o containment design limits
- o ECCS limits.

Containment design limits for accidents are based on the ASME Code requirements. The containment must be capable of accommodating the

amounts of hydrogen that may be released during a LOCA. The containment design limits are not used in the reload analysis because the ECCS limits are more restrictive. The ECCS performance limits are

- o a peak cladding temperature ≤ 2200 °F
- o a maximum cladding oxidation ≤ 17%
- o core wide metal water reaction ≤ 1%
- o the maintenance of a coolable geometry
- o assurance of long term cooling.

Only the first three limits are of significance in the reload analysis process. These three limits ensure that a short-term LOCA will preserve a coolable geometry consistent with long-term cooling requirements. Assurance of long term cooling is provided by demonstrating that the geometry is refloodable.

The fuel cladding integrity limit is used as the figure of merit for a fuel loading error. The fuel integrity limit is the MCPR at which greater than 99.9% of the fuel rod in the core would not be expected to experience boiling transition. This limit is consistent with commitments in the current plant safety analysis and represents a conservative limit for this event.

5.1.4.3 Special Events

Evaluations in section 5.2 show that four special events require reanalysis in each reload analysis:

- o shutdown margin demonstration
- o standby liquid control system capability
- o stability
- o Code overpressure protection.

Each of these events has its own limit.

The shutdown margin demonstration must provide assurance that the shutdown margin requirement in the plant technical specification will be satisfied during plant startup tests. The shutdown margin demonstration analysis assumes that the highest worth control rod is fully withdrawn.

The standby liquid control system capability analysis demonstrates that the plant can attain a cold shutdown condition without the control rods. In the analysis of this event, it is assumed that all the control rods remain at the same location or at their initial location throughout the event.

Stability evaluations must demonstrate that there will be no unsuppressed reactor power oscillations. Plant technical specifications place monitoring and operational constraints on plant operating conditions to ensure compliance with stability limits.

For overpressure protection, the limits of the ASME Code apply, which specify a peak reactor vessel pressure of 1375 psig.

5.1.5 Process for Identifying Potentially Limiting Events

In the reload analysis, it is only necessary to analyze events that can establish core operating limits, result in the need for a technical specification change, or demonstrate compliance with technical specifications. The Supply System methodology has developed a process for identifying potentially limiting events. The steps in this process are shown on Figure 5.1-2 and described below. The results of applying this process are described in section 5.2.

The process begins by identifying and eliminating all events in the current plant safety analysis that are not associated with fuel, core, or reactor system design.

The second step is to identify and eliminate all plant capability demonstrations in the current plant safety analysis. The safety analysis evaluates some special events that demonstrate plant capability to accommodate events or failure combinations not considered in the plant design basis. These events do not need to be considered in the reload analysis as long as changes in fuel and core design do not significantly alter the plant capability to accommodate these events.

The third step is to identify and eliminate all non-limiting events. The current plant safety analysis demonstrates that a number of events are not limiting. As long as the reload fuel and core design do not change the relative severity of these events, they may be eliminated from consideration in the reload analysis.

The fourth step is to identify and eliminate all events that fit within the bounding analyses in the current plant safety analysis. These events are not significantly affected by reload fuel and core design. As long as the bounding assumptions are applicable, the reload analysis does not need to analyze these events.

Any events not eliminated with these steps are considered potentially limiting. These events are the subject of the analyses documented in the reload summary report.

Table 5.1-1

RELOAD ANALYSIS LIMITS

LIMIT	VALUE
Anticipated Operational Occurrences	
Specified Acceptable Fuel Design Limits (SAFDLs)	MCPR ≥ Fuel Cladding Integrity Limit LHGR ≤ PAFF Limits
Reactor Coolant Pressure Boundary	Pressure ≤ 1375 psig
Accidents	
Reactivity Insertion	Peak Fuel Enthalpy ≤ 280 cal/gm
Emergency Core Cooling System Performance	Peak Clad Temperature ≤ 2200 °F Local Oxidation ≤ 17% Core Wide Metal Water Reaction ≤ 1%
Fuel Loading Error	MCPR ≥ Fuel Cladding Integrity Limit
Special Events	
Shutdown Margin Demonstration	Shutdown Margin ≥ Tech. Spec. Limit
Standby Liquid Control System Capability	Cold Reactor Shutdown
Stability	Suppression of Reactor Power Oscillations
Overpressure Protection	Pressure ≤ 1375 psig

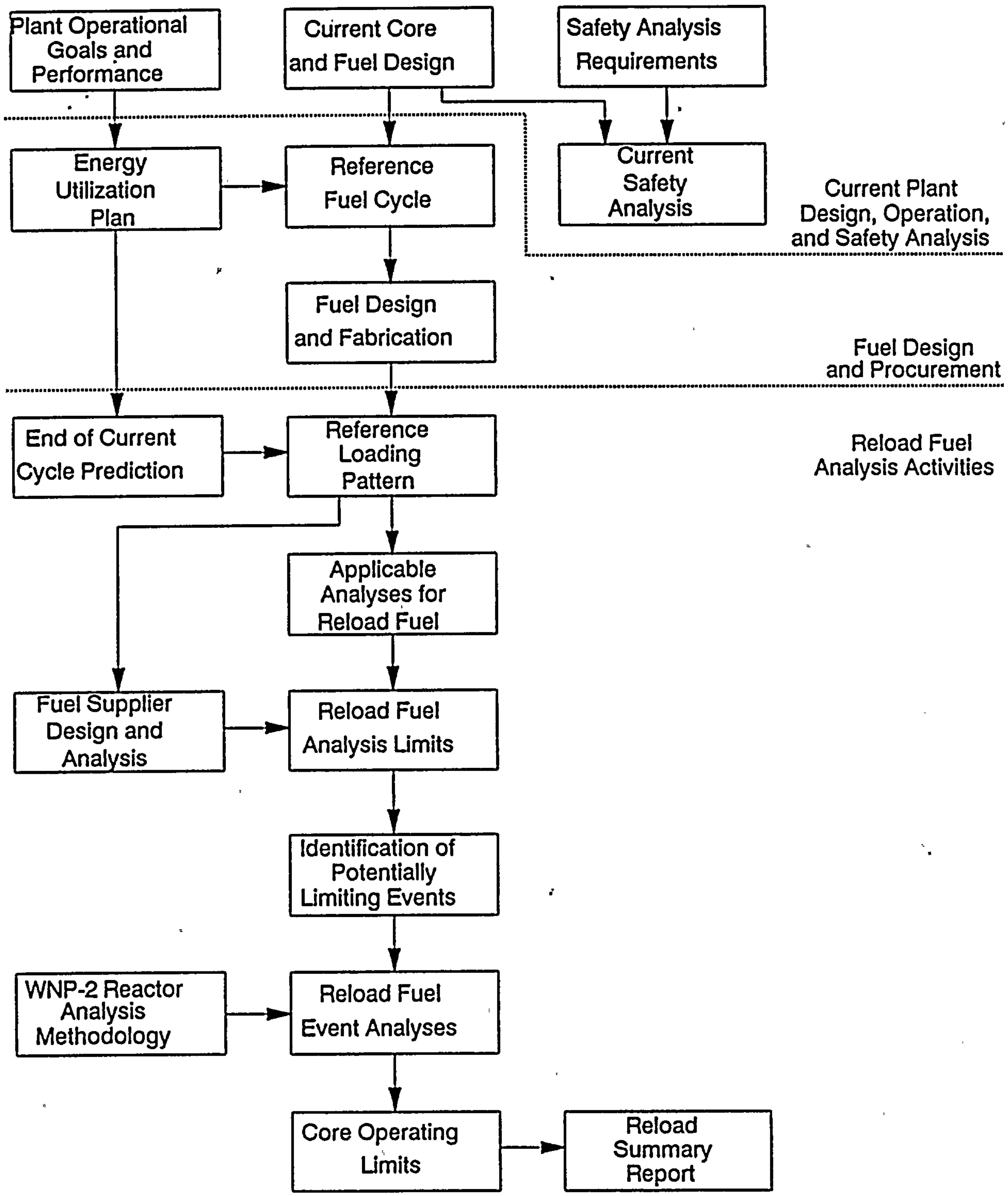

Figure 5.1-1. Reload Analysis Process Overview

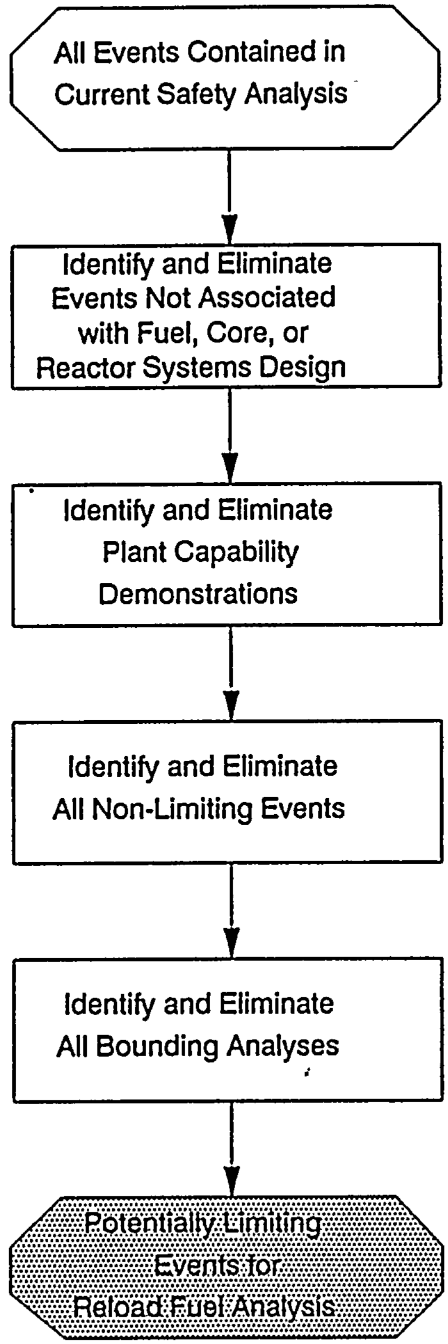

Figure 5.1-2. Process for Identifying Potentially Limiting Events

5.2 RELOAD ANALYSIS REQUIREMENTS

The reload analysis demonstrates that all plant safety analysis requirements have been satisfied for the new fuel assemblies and core configuration. Satisfying safety analysis requirements may require establishing new core operating limits or technical specifications. Reviewing the current safety analysis will ensure that necessary analyses and evaluations of the reload fuel and core configuration have been performed. The review is necessary because the reloaded core may have different nuclear and thermal hydraulic characteristics than the core configurations previously evaluated.

As noted in Section 5.1, not all events evaluated in the plant safety analysis are significantly affected by the introduction of reload fuel.

Section 5.1.5 describes the screening process used to identify potentially limiting events. The application of this process is described in more detail below. For convenience, events have are grouped by safety analysis categories:

- anticipated operational occurrences
- accidents
- special events.

Event evaluations apply only to the current plant design. Plant design changes, such as those that significantly affect system or equipment performance characteristics, may affect the relative event severity. Any such changes require assessment to determine if they affect safety analysis requirements. The Supply System assessment of plant changes will be consistent with the requirements of 10CFR50.59.

5.2.1 Anticipated Operational Occurrences Assessment

Anticipated operational occurrences are characterized by nuclear system parameter variations that pose the most significant challenge to the fuel or reactor coolant pressure boundaries. These parameter variations fall into eight categories:

- o decrease in reactor coolant temperature
- o increase in reactor pressure
- o decrease in reactor coolant flow rate
- o reactivity and power distribution anomalies
- o increase in reactor coolant inventory
- o decrease in reactor coolant inventory
- o increase in reactor coolant flow
- o increase in reactor coolant temperature.

Events in each of these categories are assessed below.

5.2.1.1 Decrease in Core Coolant Temperature

The events in this category are

- o loss of feedwater heating
- o inadvertent HPCS startup
- o inadvertent RHR shutdown cooling operation.

The current plant safety analysis demonstrates that the loss of feedwater heating is the most limiting event in this category and has the potential to establish core operating limits. Therefore, the loss of feedwater heating event is analyzed in the reload analysis.

5.2.1.2 Increase in Reactor Pressure

The events in this category are

- o DEH pressure regulator failure in the closed position

- generator load rejection
- turbine trip
- closure of the main-steam-line isolation valve
- loss of the condenser vacuum.

The current plant safety analysis demonstrates that the generator load rejection without bypass is the most limiting event in this category and has the potential to establish core operating limits. Therefore, the generator load rejection without bypass is analyzed in the reload analysis.

5.2.1.3 Decrease in Reactor Coolant Flow Rate

The events in this category are

- an RPT
- a recirculation flow control failure in the decreasing flow position.

The events in this category are relatively benign because the reduction in core flow reduces core reactivity and leads to a reduction in power prior to any system trips or actuations. The current plant safety analysis demonstrates that the events in this category are non-limiting.

5.2.1.4 Reactivity and Power Distribution Anomalies

The events in this category are

- control rod withdrawal error at low power
- control rod withdrawal error at full power.

The current plant safety analysis demonstrates that the control rod withdrawal error at full power is the most limiting event in this category and has the potential to establish core operating limits. Therefore, the control rod withdrawal error at full power is analyzed in the reload analysis.

5.2.1.5 Increase in Reactor Coolant Inventory

One event in this category is the feedwater controller failure. Other events that cause an increase in reactor coolant inventory have been discussed in sections 5.2.1.1 and 5.2.1.2. Feedwater controller failure in the maximum demand position has the potential to establish core operating limits and is therefore analyzed in the reload analysis process.

5.2.1.6 Decrease in Reactor Coolant Inventory

The events in this category are

- o inadvertent opening of the safety relief valves
- o DEH pressure regulator failure in the open position
- o loss of feedwater flow

These events are characterized by a steam and feedwater flow mismatch which results in a mild depressurization, a decrease in core power level, a decrease in water level, and a decrease in core coolant temperature prior to any system trips or actuations. The current plant safety analysis demonstrates that the events in this category are non-limiting.

5.2.1.7 Increase in Reactor Coolant Flow

The events in this category are

- o recirculation flow control failure in the increasing flow position
- o startup of an idle recirculation loop pump.

The current plant safety analysis demonstrates that the slow increase in reactor coolant flow associated with a recirculation flow controller failure may be the transient that establishes the MCPR operating limit when the reactor is operating at low power and low flow. Therefore, recirculation flow controller failure in the

increasing flow position is analyzed from low power and low flow conditions in the reload analysis.

5.2.1.8 Increase in Reactor Coolant Temperature

The one event in this category is the failure of the RHR shutdown cooling system. The current plant safety analysis demonstrates that this event is non-limiting.

5.2.1.9 Anticipated Operational Occurrences Assessment Summary

Evaluation of anticipated operational occurrences indicates that the following five events require re-evaluation during the reload analysis:

- loss of feedwater heating
- generator load rejection without bypass
- control rod withdrawal error at full power
- feedwater controller failure in the maximum demand position
- recirculation flow controller failure in the increasing flow position.

5.2.2 Accident Assessment

Accidents are postulated events that affect one or more of the radioactive material barriers. The following events are analyzed:

- control rod drop accident
- main steam line break
- instrument line pipe break
- LOCA
- fuel handling accident
- recirculation pump seizure or shaft break
- radwaste system failures
- fuel loading error.

Analyses for each of these events are described below.

5.2.2.1 Control Rod Drop Accident

The control rod drop accident bounds the consequences of reactivity insertion events in the safety analysis. In the current plant safety analysis, analysis of the consequences of a control rod drop accident are based on the results of vendor generic parametric analysis. Four input parameters to the generic analysis can change as a result of the reload fuel and core configuration. Therefore, the control rod drop accident requires reevaluation in the reload analysis process.

5.2.2.2 Main Steam Line Break

The main steam line break bounds the consequences of postulated pipe breaks outside the primary containment. The consequences of this event are based on the maximum reactor coolant activity that can exist during normal plant operation. The amount of reactor coolant activity is limited by plant technical specifications, which are not changed to reflect the reload fuel or core configuration. Therefore, the main steam line break accident does not require reevaluation in the reload analysis.

5.2.2.3 Instrument Line Break

The instrument line break bounds the consequences of pipe breaks outside the primary containment that are not automatically isolated. The consequences of this accident are based on the maximum reactor coolant activity that can exist during normal plant operation. The amount of reactor coolant activity is limited by plant technical specifications, which are not changed to reflect the reload fuel or core configuration. Therefore, the instrument line break does not require reevaluation in the reload analysis.

5.2.2.4 Loss of Coolant Accident

The LOCA bounds the consequences of pipe breaks inside the primary containment. In the current plant safety analysis, the analysis of

the consequences of a LOCA are based on the results of a conservative system analysis by the vendor. However, five input parameters to the systems analysis can be changed by the reload fuel and core configuration (see section 5.3.10). Therefore, the LOCA requires reevaluation in the reload analysis. Furthermore, introduction of a new fuel type, e.g., a new lattice design or enrichment, requires a new fuel heatup analysis. An appropriate MAPLHGR must also be established for that fuel type.

5.2.2.5 Fuel Handling Accident

The refueling accident bounds the consequences of an accident that can release radioactivity directly to the secondary containment when the primary containment is not intact. The consequences of this accident are based on an inventory of fission products in the fuel and the number of fuel rods calculated to fail as a result of the accident. These parameters are not usually expected to change significantly because of a change in the fuel or core configuration. The refueling accident will not be reanalyzed in the reload analysis unless there is a significant change in the fission product inventory of the reload fuel or in the number of fuel rods predicted to fail.

5.2.2.6 Recirculation Pump Seizure or Shaft Break

The current plant safety analysis demonstrates that a recirculation pump seizure or shaft break will not affect any core operating limits nor the basis for any technical specifications. Therefore, the recirculation pump seizure or shaft break is not evaluated in the reload analysis.

5.2.2.7 Radwaste System Failures

A conservative inventory of radwaste system radioactivity has been established for safety analysis purposes. This inventory is not affected by the reload fuel design. Therefore, radwaste system failures are not reevaluated in the reload analysis.

5.2.2.8 Fuel Loading Error

Placing a fuel assembly in the wrong location or rotating it the wrong way may impact the MCPR operating limits. Therefore, these fuel loading errors are analyzed in the reload analysis.

5.2.2.9 Accident Assessment Summary

The evaluation of accidents indicates that the following four events require examination during the reload analysis:

- control rod drop accident
- LOCA
- fuel loading error--mislocated fuel assembly
- fuel loading error--rotated fuel assembly.

5.2.3 Special Events Assessment

Special event analyses satisfy regulatory or licensing requirements not considered in the accident or anticipated operational occurrence categories. The following events are analyzed:

- shutdown margin
- standby liquid control system capability
- Code overpressure protection analysis
- stability
- shutdown from outside the control room
- anticipated transients without scram.

Analyses for these events are described below.

5.2.3.1 Shutdown Margin

The shutdown margin analysis demonstrates that the core can be made subcritical with sufficient margin when the highest-worth control rod is in the full-out position and the remaining control rods are fully inserted. For this analysis, the core is assumed to be at

cold conditions with no xenon present and at the most reactive stage in the operating cycle. Because the shutdown margin depends on the reload fuel and core design, it is evaluated in the reload analysis.

5.2.3.2 Standby Liquid Control System Capability

The standby liquid control system capability analysis demonstrates that the core can be made subcritical by actuating the standby liquid control system. The analysis verifies that the core becomes subcritical at all power conditions with minimum control rod inventory and no xenon present. Because the standby liquid control system capability depends on the reload fuel and core design, it is analyzed in the reload analysis.

5.2.3.3 Code Overpressure Protection Analysis

The Code overpressure protection analysis demonstrates conformance to the ASME Code requirement for demonstrating protection against overpressure. The reload fuel and core design affect overpressure protection. Therefore, the Code overpressure protection analysis is performed for each reload analysis.

5.2.3.4 Stability

The stability evaluations demonstrate that no divergent power oscillations will occur that cannot be detected and suppressed. Stability margins depend on nuclear and thermal hydraulic parameters that change with each reload. Plant technical specifications, however, contain requirements that restrict plant operation within the power/flow map. These restrictions are consistent with the General Electric Company recommendations in Service Information Letter (SIL) 380. Furthermore, WNP-2 has a stability monitoring system. Technical specification action statements are provided to assure compliance with the event acceptance limits associated with stability. Therefore, stability

is not evaluated in the reload analysis as long as the technical specification bases are applicable.

5.2.3.5 Shutdown from Outside the Control Room

The evaluation of shutdown from outside the control room ensures that the plant can be shut down with equipment outside the control room. This capability is not significantly impacted by the reload fuel and core design. Therefore, shutdown from outside the control room is not evaluated in the reload analysis.

5.2.3.6 Anticipated Transients Without Scram

An anticipated transient without a scram is a postulated transient that reaches a reactor protection system (scram) setpoint or requires a manual scram without insertable control rods to insert to shut the reactor down. This capability has been evaluated in the current plant safety analysis using conservative analysis assumptions. Therefore, anticipated transients without a scram are not evaluated in the reload analysis.

5.2.3.7 Special Events Assessment Summary

The evaluation of special events indicates that the following four events require examination during the reload analysis:

- o shutdown margin
- o standby liquid control system capability
- o Code overpressure protection analysis
- o stability.

5.2.4 Potentially Limiting Events for Reload Analysis

The screening process indicates that fourteen events require reevaluation in each reload analysis. Table 5.2-1 lists these events and their event category.

Table 5.2-1

Events Requiring Reevaluation in the Reload Analysis
(Arranged by Category)

Anticipated Operational Occurrences

Loss of Feedwater Heating

Generator Load Rejection without Bypass

Control Rod Withdrawal Error

Feedwater Controller Failure - Maximum Demand

Recirculation Flow Controller Failure - Increasing Flow

Accidents

Control Rod Drop Accident

Loss of Coolant Accident

Fuel Loading Error - Mislocated Fuel Assembly

Fuel Loading Error - Rotated Fuel Assembly

Special Events

Shutdown Margin Demonstration

Standby Liquid Control System Capability

Code Overpressure Protection Analysis

Stability.

5.3 EVENT ANALYSIS PROCEDURES

The categorization of events in section 5.2 is consistent with recommendations in the EPRI guidelines for BWR event analysis [Reference 20].

The event analysis procedures establish the requirements for specific event analyses. Section 5.3.1 provides an overview of the procedures. Sections 5.3.2 through 5.3.13 summarize specific event analysis procedures and the bases for the procedures.

5.3.1 Event Analysis Overview

The Supply System event analysis methodology is conservative to ensure that analysis uncertainties are not underestimated. Conservatism in the analysis process is imposed by the way inputs are selected and uncertainties are treated.

The description of each event analysis procedure follows an outline, which ensures consistent coverage and a comprehensive basis for development. Each description provides an understanding of the assumptions and phenomena that can impact analysis results significantly if they are changed by the reload fuel and core design. Each procedure description contains the following:

- an event description, including initial conditions, operational assumptions, required operator response, and event acceptance limits
- analysis considerations, including key phenomena, systems considerations, and component performance characteristics
- a discussion of the methodology and computer codes used in the event analysis
- steps in the licensing analysis procedure
- a description of the sensitivity studies used to justify the procedure
- typical analysis results.

The scope and content of each item are described below.

5.3.1.1 Event Description

The event description provides an overview of the event sequence from its starting point until a stable condition is reached. It ensures that the event is appropriately simulated. The event description discusses the response of the plant systems to the initiating event, including the effects of all assumed system trips or initiations. The event description specifies the initial conditions assumed for the event, system operational considerations, the anticipated operator response to the event, and event acceptance limits.

The initial conditions define the plant operating state used as the starting point for the event analysis. Operational considerations identify the plant operating mode used in the analysis, any additional failures that must be considered in the analysis, and system trips and equipment initiations that will probably be encountered during the event.

Operator response describes the actions that will be taken by plant operators to attain a stable operating state. Operator response should not be considered a viable option unless information and time are sufficient for the necessary tasks.

Event acceptance limits identify the figures of merit used to demonstrate the acceptability of event consequences. The event acceptance limits are consistent with the reload analysis limits listed in Table 5.1-1. This table lists only limits specifically challenged by the event identified.

5.3.1.2 Analysis Consideration

Many factors may impact event analyses. These include

- o key event phenomena

- systems performance considerations
- component performance characteristics.

The importance of each of these is discussed below.

In event analysis, it is important to understand the phenomena that have a significant impact on the event. Different approaches and different modelling options may be used to simulate the phenomena of interest; which approach is chosen depends in part on the relative importance of each to the event.

The interaction of plant systems may have a significant impact on an event and its consequences. Systems considerations identify all systems that may significantly impact event analyses; these system require simulation in the event analysis.

The performance characteristics of individual components may also significantly impact event consequences. Component performance characteristics include instrument setpoints, valve opening and closing times, instrumentation delay times, and pump coastdown characteristics.

5.3.1.3 Methodology/Integration of Codes and Analysis

The Supply System methodology uses an integrated set of computer codes and a variety of analysis techniques, such as SCU methods. The codes and techniques are described in section 2.0.

5.3.1.4 Licensing Analysis Procedure

The licensing analysis procedure identifies key assumptions and modelling considerations used in the event analyses. Included are assumptions about the plant operating state and the system and component simulations that must be included.

5.3.1.5 Sensitivity Studies/Justification of Procedure

This section identifies the sensitivity studies that provide the basis for the licensing analysis procedure. It also discusses the results of these studies. The studies typically consider uncertainties associated with event definition, the plant operating state, instrumentation that measures specific parameters or initiates protective action, and analytical models.

5.3.1.6 Typical Results

This section describes the results of applying the reload analysis methodology to WNP-2 Cycle 4. The analysis results are compared with results from the fuel vendor analysis for Cycle 4. This allows a comparison of the results obtained using the Supply System methodology with the results obtained using an approved methodology.

5.3.2 Loss of Feedwater Heating

5.3.2.1 Event Description

Loss of feedwater heating results in a core power increase and power distribution shift due to an increase in core inlet subcooling. Feedwater heating can be lost in at least two ways:

- o a steam extraction line to one of the heaters is closed
- o feedwater flow bypasses one or more feedwaterheaters.

The first case produces a gradual cooling of the feedwater. In the second case, the feedwater bypasses the heaters and no heating of the bypass feedwater occurs. In either case, the reactor vessel receives cooler feedwater. The maximum number of feedwater heaters that can be tripped or bypassed by a single event represents the most severe situation.

Because of the mixing of feedwater and recirculation flow, the rapid decrease in feedwater temperature results in a gradual increase in core inlet subcooling. This causes a relatively slow power increase and shift in the power distribution toward the bottom of the core. As a result of the core power increase, the vessel steam flow increases. This results in an increase in nuclear system pressure due to the larger steam line pressure drops, assuming that the pressure regulator acts to maintain constant turbine inlet pressure. A scram on average power range monitor (APRM) high neutron flux or APRM thermal power may occur depending on the magnitude of the power increase. The increase in inlet subcooling tends to mitigate the decrease of the MCPR caused by the core power increase.

Table 5.3.2-1 shows the expected sequence of events for the loss of feedwater heating.

5.3.2.1.1 Initial Conditions and Operational Assumptions

The following plant operational conditions and assumptions form the principal bases for the analysis of the loss of feedwater heating event:

- o The plant is operating at rated core power and rated core flow.
- o The remaining NSSS operating parameters are consistent with normal plant operation.
- o The maximum change in feedwater temperature identified in the plant safety analysis is assumed to occur instantaneously.
- o The plant is operating in the manual flow control mode.

5.3.2.1.2 Operator Actions

If the core power increase does not cause reactor scram, the reactor will settle out at a higher steady state power level. An APRM neutron flux or thermal power monitor alarm will alert the operator to insert control rods or reduce core flow to get back into the normal range of the power/flow operating map. The operator must determine, from existing tables, the maximum allowable turbine-generator output with feedwater heaters out of service. If reactor scram occurs, the operator must monitor the reactor water level and pressure controls and the turbine generator auxiliaries during coastdown.

5.3.2.1.3 Event Acceptance Limits

The acceptance limits for this event are a MCPR ≥ fuel cladding integrity limit and the LHGR ≤ PAFF limits. Compliance to the fuel cladding integrity limit is demonstrated by assuring that the operating limit MCPR is greater than or equal to the MCPR which assures that greater than 99.9% of the fuel rods in the core are

not expected to experience boiling transition (safety limit) plus the change in ΔCPR during the event. Compliance to the PAFF limits is demonstrated by assuring that the change in fuel design limit ratio (ΔFDLRX) is less than 0.2 where FDLRX is the ratio of LHGR to the LHGR limit.

5.3.2.2 Analysis Considerations

This section describes the key analysis considerations applicable to the loss of feedwater heating event. It includes: (1) a description of the phenomena occurring during the event that have a significant impact on the event consequences; (2) a discussion of the system performance characteristics that can significantly affect the course of the event; and (3) a discussion of the performance characteristics of the important components as they relate to the event consequences.

5.3.2.2.1 Key Phenomena

Described below are the key phenomena related to the loss of feedwater heating event. Consideration of these phenomena is necessary in the simulation of this event to accurately model the plant response.

Feedwater Phenomena As the colder feedwater flow is mixed with the recirculation flow, the core inlet enthalpy decreases and core power increases. The feedwater flow increases slightly to maintain the reactor water level.

Recirculation Phenomena In this analysis the recirculation flow rate is held constant.

Pressure Vessel Phenomena As the colder feedwater enters the reactor pressure vessel, it mixes with the recirculation flow in the downcomer region. Further mixing occurs in the vessel lower plenum. This mixing phenomena determines the core inlet

temperature corresponding to a change in feedwater temperature. The vessel pressure increases due to the increase in core power.

Core Phenomena The reduction of core inlet temperature and the corresponding reduction of the core average void fraction causes a positive reactivity insertion. This causes an increase in power level which increases core heat flux and generates additional steam voids. If the core power level does not reach the pre-established scram setpoint, the increased steam voids and Doppler reactivity will limit the power increase and a new equilibrium power level will be established. If scram occurs, the power increase will be rapidly terminated.

5.3.2.2.2 Systems Considerations

The event is initiated by a rapid reduction of feedwater temperature resulting from the limiting single failure in the feedwater or steam extraction systems. All other systems are assumed to operate as designed and to continue to function throughout the event. Safety systems are assumed to initiate at their pre-established setpoints. The important systems are: (1) the core and fuel system, including the nuclear and thermal hydraulic coupling; (2) the steam and feedwater systems; (3) the recirculation system; and (4) the reactor protection and scram systems, including the APRM flux or thermal power scram. Other systems which may be initiated as a result of a scram or operator action are not required because they have no effect until after the challenge or nearest approach to SAFDLs has occurred.

5.3.2.2.3 Component Performance Characteristics

Modeling of the NSSS components outside of the reactor core is simplistic because of the slow nature of the transient. The core response is calculated at a steady state condition to assure that the core inlet enthalpy and its maximal effects on core power and power distribution are accounted for.

5.3.2.3 Methodology/Integration of Codes and Analysis

The loss of feedwater heating event is simulated with the use of reactor physics codes (lattice physics and three-dimensional simulator) by calculating the core response at a steady state condition at the event terminating point (i.e., thermal equilibrium consistent with the maximum feedwater temperature change). This method provides an accurate calculation of the core power level and enables an assessment of the radial and axial power distribution shifts that occur.

Analysis of the loss of feedwater heating event utilizes the SIMULATE-E three-dimensional BWR simulator code for calculation of core response to a reduction in inlet temperature. The lattice physics input to SIMULATE-E is provided by CASMO-2E. The MICBURN-E code is used to determine the gadolinia cross sections used in CASMO-2E and ESCORE provides the fuel temperature distribution. (See section 2 and figure 2-1 for an overview of the overall WNP-2 reactor analysis methodology computer code sequence.)

SIMULATE-E calculates the reactor power, power distribution, and fuel assembly flow rates as a function of core inlet temperature and core flow. The SIMULATE-E analysis represents the reactor in equilibrium for the initial conditions and after the 100°F change in feedwater temperature. TLIM uses the results of the SIMULATE-E cases to calculate the ΔCPR and the LHGR during the event for comparison to the event acceptance limits.

Initial conditions are developed from the operating statepoints predicted in the cycle step through analysis. Final conditions are determined by reducing the feedwater temperature by 100°F and increasing the core power such that the calculated eigenvalue remains unchanged. The change in feedwater temperature is modeled by increasing the core inlet subcooling since feedwater temperature is not specified in SIMULATE-E input.

5.3.2.4 Licensing Analysis Procedure

The loss of feedwater heating transient is analyzed assuming a 100°F decrease in the feedwater temperature caused by the failure of plant feedwater heaters. The plant is designed such that the maximum change in feedwater temperature, considering the limiting single failure in the feedwater system, is less than 100°F. Therefore, this transient disturbance is treated as an anticipated operational occurrence. Based on plant evaluations and startup test results, the maximum change in feedwater heating for the limiting single failure is approximately 60°F. These evaluations and test results demonstrate the conservatism inherent in the selection of the 100°F loss of feedwater heating used in the licensing analysis procedure.

The loss of feedwater heating licensing analysis is initiated from rated power and flow conditions (100/100). The analysis is based on beginning of cycle conditions (~0.2 GWD/MTU). Rated feedwater temperature (414°F) is assumed at the beginning of the event and the feedwater temperature is assumed to be instantaneously reduced by 100°F (to 314°F) during the event. For operating cycles in which final feedwater temperature reduction is to be employed to extend the operating cycle, the end of cycle all rods out full power conditions will be analyzed assuming a 100°F drop in feedwater temperature from the minimum feedwater temperature to be utilized in this mode of operation.

In the analysis of the loss of feedwater heating event, two SIMULATE-E cases are run. The first case represents the initial conditions and the second case represents the post event steady state conditions. In the licensing analysis procedure, any scram setpoints reached as a result of the event are conservatively ignored.

Based on the sensitivity studies performed for the loss of feedwater heating (section 5.3.2.5.2), this event is not expected

to be limiting in the fuel reload analysis process. Furthermore, the conservatism inherent in the event definition are expected to bound any uncertainties in the analysis process. However, the sensitivity studies do indicate that there may be a small non-conservatism introduced by the assumption of beginning of cycle exposure condition, the assumption of constant pressure during the event, and the use of the rated power/flow conditions. If the ΔCPR calculated using the licensing analysis procedure is within a ΔCPR of 0.02 of being the limiting event or if the ΔFDLRX is greater than 0.18, then the limiting exposure point will be evaluated with an increase in core pressure consistent with the change in reactor operating state for the limiting power/flow condition bounded the maximum power level required to be used in the safety analysis process.

5.3.2.5 Sensitivity Studies/Justification of Procedure

This section describes the sensitivity studies performed to provide the basis for the licensing analysis procedure. The sensitivity studies are intended to cover the primary uncertainties related to the loss of feedwater heating event, including: (1) operating state; (2) instrumentation; (3) methodology; and (4) event definition. These sources of uncertainty, as related to this event, are described in more detail below.

5.3.2.5.1 Primary Uncertainties

Operating State Uncertainties include both fuel assembly and system uncertainties. System uncertainties are those related to feedwater flow and temperature, the TIP system, core flow rate, core pressure, core/assembly power and power distribution, etc. Fuel assembly uncertainties are those related to individual fuel assemblies, such as local peaking factor, assembly flow rate, heat transfer correlations, etc.

In the reload analysis process, the magnitudes of the operating state uncertainties are not expected to change during the event.

In the short period of time between initiation of the event and a return to reactor equilibrium, it is not expected that the error in any plant parameter would significantly change. Thus, if the reactor dome pressure reading is 5 psia higher than actual at initiation of the event, it is assumed to be 5 psia high at the termination of the event. With essentially no change in the uncertainty of the plant parameters during the loss of feedwater heating event, the primary approach is to account for the uncertainty in the plant operating state assumed in the event analysis. The uncertainties in the plant operating state are included in the event acceptance limit.

Instrumentation Uncertainties are the uncertainties in the instrument system or sensor performance associated with the specific value at which an instrument trip signal is generated. In the analysis of the loss of feedwater heating event, no protection system intervention is assumed to occur (high neutron flux and thermal power trips are conservatively ignored). Therefore, no consideration of instrumentation uncertainties is necessary.

Methodology Related Uncertainties are associated with the technique used to analyze this event. The licensing analysis methodology uses SIMULATE-E to calculate the change in reactor parameters. The major uncertainty in this method is in using a steady state simulation to analyze this transient. Because the loss of feedwater heating event takes place very slowly, the core remains in a quasi steady state condition. The use of a steady state code such as SIMULATE-E is therefore reasonable. The conservatism of introducing into the analysis process by the event definition, as described below, bounds the effect of the steady state analysis.

Event Definition Uncertainties are introduced through the assumption of the event scenario. For the loss of feedwater heating event, the primary conservatisms and uncertainties are introduced by: the assumption of the magnitude of the temperature reduction; the use of beginning of cycle exposure; and the assumption of constant reactor pressure. Each of these parameters is discussed

in more detail below. Other uncertainties introduced by the use of rated power and flow as the event initial conditions have been covered in the event sensitivity analyses (section 5.3.2.5.2).

The major conservatism in the initial conditions is the magnitude of the feedwater temperature decrease. The loss of feedwater heating analysis methodology assumes an instantaneous 100°F feedwater temperature decrease. Operating experience indicates that the temperature change in the majority of loss of feedwater heating events is significantly less. The relatively few events that have occurred and which approach the magnitude assumed in the license basis analysis have been terminated by operator action or protection system operation before any design limits where challenged. The maximum drop in feedwater temperature due to any single failure for WNP-2 has been evaluated to be about 34°F. Based on the results of the sensitivity analyses (section 5.3.2.5.2) for a 60°F feedwater temperature drop, the analysis has about 0.03 ΔCPR conservatism relative to the assumed worst case (100°F) feedwater heater failure. The 60°F loss of feedwater heating is considered to be a more reasonable maximum bound on the WNP-2 feedwater system design and operation. Therefore, the use of an instantaneous reduction of 100°F in feedwater temperature represents a significant conservatism in the event analysis.

The base case for the loss of feedwater heating is the beginning of cycle exposure (about 0.2 GWD/MTU). The reactor kinetics parameters change as the core exposure increases. Sensitivity studies of the exposure effects on the event analysis results were performed and are described in section 5.3.2.5.2. Based on the results of these sensitivity studies, it has been determined that a slight nonconservatism (about 0.01 ΔCPR) may be introduced by the assumption of beginning of cycle exposure. This source of potential nonconservatism has been considered in the development of the licensing analysis process (section 5.3.2.4).

The analysis of the loss of feedwater heating event assumes constant reactor pressure throughout the event. Because of the slight increase in core power during the event, a slight increase in core pressure could result. A small increase in core pressure will cause a small increase in calculated ΔCPR. This potential nonconservatism has been considered in the development of the licensing analysis procedure.

5.3.2.5.2 Event Sensitivity Analysis Results

Sensitivity analyses were performed to evaluate the magnitude of the effect of the principal conservatism and uncertainties associated with the loss of feedwater heating event. The sensitivity studies cover the cycle exposure effects, the magnitude of the feedwater temperature reduction, and the potential for operating at conditions other than rated.

Tables 5.3.2-2 provides a summary of the 29 cases run to support the licensing analysis procedure for the loss of feedwater heating event.

The first 20 cases were run using the control rod step through for Cycle 4 at each exposure point. Cases 21-29 were run to quantify conservatism in the method, analyze off rated conditions, and determine the effect of the core power distribution shift.

Cases 1-20 analyze the loss of feedwater heating event at each of ten exposure points in the cycle step through. Two cases are run at each exposure point. The first case is the standard step through case at the exposure point of interest. The second case in the pair is run with the inlet subcooling increased to simulate a 100°F drop in feedwater temperature. The eigenvalue from the first case in each pair of runs is saved and in the second case, SIMULATE-E is instructed to converge on the eigenvalue from the first case. This sequence of cases then simulates the reactor in a critical state before and after the loss of feedwater heating

event. The ΔCPR and change in LHGR is determined at each exposure point. For the limiting exposure point, the ΔCPR for the event is less than 0.01 greater than for beginning of cycle, and the ΔFDLRX is less than 8% greater than for beginning of cycle.

The nine cases run in addition to the exposure sensitivity analysis cases were to determine the effects of the initial condition assumptions and event definition in the method. These cases are described below.

Cases 21-22 were run to determine the effect of the change in the axial power distribution during the event. To assess the effect of the change in axial power distribution, case 22 was run using the same initial conditions as case 21, but the axial power distribution was constrained to remain constant throughout the event. The results indicate that holding the power distribution constant gives a less than 0.020 higher ΔCPR for the event. The constant power assumption for this event is not realistic because of the phenomena involved, primarily because of the power shift that occurs due to the change in core inlet subcooling. However, this case demonstrates that the overall effect of the change in axial power distribution has a relatively insignificant effect on ΔCPR. Furthermore, the use of SIMULATE-E in the analysis process more accurately simulates the actual phenomena involved.

Case 23 was run to quantify the effect of a smaller drop in feedwater temperature. Based on the design of the Number 6 heaters in the feedwater system, a drop of 60°F would be expected if both number 6 heaters were lost. Due to feedback mechanisms associated with the extraction system on the heaters, the actual drop in feedwater temperature is less than 60°F when both of these heaters are lost. For the purposes of this analysis, a 60°F drop in feedwater temperature was analyzed. The 60°F change bounds the 34°F drop if both Number 6 heaters were lost or if an entire string were to fail. Results of the 60°F drop in feedwater temperature show that the ΔCPR is reduced by about 0.02 compared to the base case

and a corresponding reduction in ΔFDLRX. This case demonstrates that the assumed change in feedwater temperature during the event adds a significant amount of conservatism to the results.

Cases 24 through 29 were all done at off rated power/flow conditions on the WNP-2 power/flow map. Case 24-25 are a pair of runs that analyze the event at 100% power and 87% flow. Cases 26-27 analyze the 100% power, 105% flow point. Cases 28-29 are for 104.2% power and 100% flow. In all three of these pairs of runs, there is very little change in the ΔCPR and the resulting ΔCPR is bounded by the 100% power, 100% flow result. The maximum increase in ΔFDLRX over the base case was about 17%. These cases demonstrate that the ΔCPR due to the loss of feedwater heating is relatively insensitive to the selection of the initial power and flow conditions.

5.3.2.6 Typical Results

The licensing analysis procedure for evaluating the loss of feedwater heating event is consistent with the fuel supplier (ANF) methods and gives similar results. Cases 1 and 2 on Table 5.3.2-2 represent the WNP-2 reactor analysis methodology using the licensing analysis procedure for the loss of feedwater heating for Cycle 4. Based on this methodology, the ΔCPR would be reported as 0.07. The core wide MCPR at beginning of cycle is 1.679. After a 100°F temperature reduction, the core wide MCPR is 1.583. This is a change of corewide MCPR of 0.096. From the reload report supplied by ANF for Cycle 4, the result calculated is 0.09. Therefore, it is concluded that the Supply System licensing analysis procedure is reasonable yet conservative based on the evaluation of the analysis conservatisms and uncertainties and the comparison with the ANF results.

Table 5.3.2-1

Sequence of Events for Loss of Feedwater Heating

Maximum feedwater temperature reduction is assumed to occur instantaneously.

Initial effect of reduced temperature feedwater starts to raise core power level and steam flow.

Turbine control valves open to control pressure.

APRM or thermal power alarm setpoint reached.

If the core power does not reach the scram setpoint, new steady state operating condition is achieved.

If core power reaches scram setpoint, the APRMs will initiate a reactor scram. The NSSS will follow the same sequence as for a scram from normal power level.

Table 5.3.2-2

Results of Loss of Feedwater Heating Analysis

Cases	Initial Power MW	Flow Mlb/hr	Exposure GWD/MTU	Delta FW Temp °F	Final Power MW	Delta FDLRX	Delta MCPR
1-2	3323	108.5	0.196	100.0	3622	0.111	0.064
3-4	3323	108.5	0.6	100.0	3616	0.119	0.047
5-6	3323	108.5	1.2	100.0	3607	0.114	0.053
7-8	3323	108.5	1.8	100.0	3620	0.114	0.067
9-10	3323	108.5	2.4	100.0	3619	0.111	0.056
11-12	3323	108.5	3.0	100.0	3614	0.111	0.051
13-14	3323	108.5	3.6	100.0	3616	0.099	0.048
15-16	3323	108.5	4.2	100.0	3625	0.043	0.055
17-18	3323	108.5	4.8	100.0	3610	0.049	0.054
19-20	3323	113.9	5.505	100.0	3619	0.084	0.057
21-22	3323	108.5	0.196	100.0	3679	N/A	0.079
23	3323	108.5	0.196	60.0	3512	0.075	0.042
24-25	3323	94.4	0.196	100.0	3617	0.131	0.055
26-27	3323	113.9	0.196	100.0	3617	0.115	0.059
28-29	3473	108.5	0.196	100.0	3782	0.130	0.061

5.3.3 Generator Load Rejection Without Bypass

5.3.3.1 Event Description

The generator load rejection without bypass (LRNB) event is the postulated complete loss of electrical load to the turbine generator coupled with the assumed failure of the turbine bypass system. Fast closure of the turbine control valves is initiated whenever electrical grid disturbances occur which result in significant loss of load on the generator. The turbine control valves are required to close rapidly to prevent overspeed of the turbine generator rotor due to the loss of load. The rapid closure of the turbine control valves causes a sudden reduction of steam flow which results in a nuclear system pressure increase. Neutron flux increases rapidly because of the core void reduction caused by the pressure increase. Turbine control valve fast closure initiates a scram trip signal and a prompt RPT, which results in a rapid reactor shutdown. The reactor vessel pressure increase is limited by the action of the relief valves. The neutron flux increase is limited by the scram and the prompt RPT. The peak fuel surface heat flux increases initially due to the neutron flux increase then decreases following reactor shutdown. Long term reactor water makeup is provided by the feedwater system or high pressure makeup systems. Heat rejection is through the relief valves to the suppression pool.

Table 5.3.3-1 shows the expected sequence of events for the generator load rejection without bypass transient.

5.3.3.1.1 Initial Conditions and Operational Assumptions

The following plant operational conditions and assumptions form the principal bases for the analysis of the generator load rejection without bypass event:

- o The plant is operating at the safety analysis power level and rated core flow.

- The remaining NSSS operating parameters are consistent with normal plant operation.
- A generator load rejection initiates the transient which results in a control valve fast closure at the fastest design rate.
- A reactor scram is initiated by the control valve fast closure.
- The pressure relief function is available to limit the pressure increase.
- The bypass system is assumed to fail in the closed position.
- All of the remaining plant control systems function normally.
- The system trips and initiation signals are consistent with the technical specifications.
- The prompt RPT system is initiated by the control valve fast closure and trips both recirculation pumps.

5.3.3.1.2 Operator Actions

No restart is assumed and the reactor is to be cooled down. The operator is expected to take the following actions as appropriate:

- Control the reactor pressure.
- Ascertain that all control rods are in.
- Monitor and maintain reactor water level
- Cool down the reactor consistent with plant procedures.

5.3.3.1.3 Event Acceptance Limits

The acceptance limits for this event are MCPR ≥ fuel cladding integrity limit; LHGR ≤ PAFF limits; and reactor pressure ≤ the ASME Code limit for the reactor coolant pressure boundary. Compliance to the fuel cladding integrity limit is demonstrated by assuring that the operating limit MCPR is greater than or equal to the fuel cladding integrity MCPR limit (which assures that greater than 99.9% of the fuel rods in the core are not expected to

experience boiling transition) plus the change in ΔCPR during the event. Compliance to the protection against the PAFF limit is assured by meeting the LHGR limit requirements for transient occurences in the fuel vendor mechanical design topical reports (ex. Reference 16). Compliance with the ASME Code limit for the reactor coolant pressure boundary is demonstrated by assuring that the peak reactor vessel pressure is less than 1375 psig.

5.3.3.2 Analysis Considerations

This section describes the key analysis considerations applicable to the generator load rejection without bypass event. It includes: (1) a description of the phenomena occurring during the event that have a significant impact on the event consequences; (2) a discussion of the system performance characteristics that can significantly affect the course of the event; and (3) a discussion of the performance characteristics of the important components as they relate to the event consequences.

5.3.3.2.1 Key Phenomena

Described below are the key phenomena related to the generator load rejection without bypass event. Consideration of these phenomena is necessary in the simulation of this event to accurately model the plant response.

The generator load rejection without bypass involves the reactor core, the entire reactor coolant pressure boundary, and the main steam system. The event is characterized by rapidly changing conditions with complex interactions associated with the key phenomena.

<u>Steam Line Phenomena</u> The event begins with closure of the turbine control valves which causes a pressure increase at the turbine inlet that is rapidly transmitted to the reactor pressure vessel by pressure wave phenomena in the steam lines. The relief valves open at pre-established setpoints allowing a steam release path for

pressure relief. Nodalization of the steam lines is necessary to assure accurate simulation of the system pressure response.

Pressure Vessel Phenomena The propagation of the pressure wave from the steam lines to the core is an important phenomenon. The attenuation of the pressure wave by the reactor vessel internal components is a particularly important phenomenon in modeling the timing of the core moderator void changes.

Recirculation Phenomena Following the generator load rejection, a turbine generator overspeed will occur which is limited by the fast closure of the turbine control valves.

Modeling of the recirculation flow coastdown phenomena following RPT is important to assure that the changing recirculation flow is correctly calculated. The recirculation system modeling includes consideration of the downcomer phenomena, including the annular flow region above and through the jet pumps. The changing performance of the jet pumps at varying pressure and drive flow is included. The external recirculation loop flow is represented so that flow into the vessel as a function of time is accurately simulated.

Core Phenomena The phenomena important in the core region are the reactivity effects that contribute to changes in neutron flux level and hence energy generation and power input to the coolant. The primary reactivity feedback effects are steam void reactivity, fuel temperature Doppler reactivity, and control rod reactivity. The steam void reactivity contributes to the initial positive reactivity as a result of steam void collapse as the pressure increase from the steam system reaches the core.

During the generator load rejection transient, the collapsing moderator voids cause an increase in power level which in turn increases fuel temperature and moderator voids to the point that power would reach a new level. However, with steam flow restricted, system pressure and core power continue to rise until

a rapid scram of control rods and trip of recirculation pumps initiated by the turbine control valve fast closure terminate the power increase and result in reactor shutdown.

5.3.3.2.2 Systems Considerations

For the generator load rejection transient, the initiating event is the action that causes the generator load rejection to occur. All other systems normally operating are assumed to function as designed. Safety systems that are designed to actuate are assumed to actuate at their pre-established setpoints. The steam bypass system is assumed not to function.

The important systems to be considered are: (1) the reactor protection system including the turbine control valve fast closure scram; (2) the control rod drive (scram) system; (3) the steam system including control valves and relief valves; (4) the recirculation system, including the prompt RPT; (5) the steam separation system inside the vessel; and (6) the fuel and core system, including the nuclear and thermal hydraulic coupling. Other systems called upon for long term operation are not required to be part of this analysis because their action occurs much later in the transient, following the time of challenge or nearest approach to the event acceptance limits.

5.3.3.2.3 Component Performance Characteristics

The generator load rejection transient analysis requires detailed modeling of the NSSS in order to assure that all systems that influence reactor system pressure, steam flow, core flow, and core inlet enthalpy are properly considered. The selection of conservative or licensing basis component performance characteristics is based on a buildup of conservative assumptions established by past practices and licensing requirements.

<u>Turbine Control Valve Closure Characteristics</u> The turbine control valve closing characteristics are fundamental to the generator load

rejection transient. The turbine control valve closure characteristics are specified by the turbine manufacturer and are designed to provide adequate protection to the turbine from potential overspeed conditions. The closing characteristics are designed to assure rapid closure and are verified during plant startup testing. The fastest specified closure time is used in the analysis. The turbine control valve closure is simulated as linear from its actual operating position to fully closed.

Relief Valve Characteristics The relief valves are used to protect the reactor coolant pressure boundary against overpressure events. The technical or design specifications establish limiting conditions for the relief valve setpoints. The maximum values are used in the licensing basis analysis in order to assure a conservative evaluation of the system pressure response.

Recirculation Pump Coastdown Characteristics The recirculation pumps receive a trip signal from the same source as the turbine control valve fast closure scram signal which opens breakers to the recirculation pumps power supply. This provides additional negative moderator void reactivity insertion to reduce the magnitude of the transient neutron flux rise. The rate of reactivity insertion is strongly influenced by the timing of pump coastdown which directly influences the amount of core recirculation flow driven through the jet pumps. The breaker time, in terms of number of cycles to trip power, and the inertia of the pumps are specified in plant design and confirmed during plant startup testing. The slowest pump coastdown consistent with design specifications is used in the analysis. In addition, the maximum instrumentation and signal processing delay times are used in the analysis.

Turbine Control Valve Fast Closure Scram Signal Scram on turbine control valve fast closure is assumed in the generator load rejection analysis. The turbine control valve fast closure scram signal is generated from hydraulic oil pressure associated with the disc dump valve. The maximum design specification time between the

start of the turbine control valve fast closure and the hydraulic oil pressure switch actuation is used in the analysis.

Reactor Protection System Signal Delays The reactor protection system includes the collection of a number of analog and digital signals, conditioning of these signals, comparison to pre-established setpoint limits, and activation of nuclear system trips. The signal processing and trip initiation involves delay times which impact transient response. The plant technical specifications identify the allowable reactor protection system response times and are used in the analysis.

Control Rod Drive Insertion Time The control rod drive system provides the primary mechanism for negative reactivity insertion for terminating the transient. The control rod drives are inserted in the scram mode by the control rod hydraulic control system. The control rod scram time is determined from surveillance test data.

Pressure Switches in the Hydraulic Oil System The pressure switches in the hydraulic oil system are assumed to initiate a reactor scram for this event. The design specifications that maximize the scram initiation time are used in the analysis. The specific values for the parameters associated with the component performance in the licensing basis model are given in Table 4-1 in Reference 9. The table also compares the values for the licensing and nominal conditions.

5.3.3.3 Methodology/Integration of Codes and Analysis

The primary analysis model in the simulation of the generator load rejection without bypass event is the system thermal hydraulic model, RETRAN-02. RETRAN-02 is used to calculate the changes in system and core average nuclear and thermal hydraulic parameters throughout the course of the event. The RETRAN-02 analysis results are used in the assessment of fuel thermal margin, the increase in nodal power, and the peak reactor vessel pressure.

The analysis of the generator load rejection without bypass is performed using the following codes in the sequence shown on Figure 2-1: (1) ESCORE; (2) MICBURN-E; (3) CASMO-2E; (4) NORGE-B; (5) SIMULATE-E; (6) SIMTRAN-E; (7) STRODE; (8) RETRAN-02; (9) FICE; (10) VIPRE-01; and (11) STARS (when limiting). ESCORE is used to provide the fuel rod temperature distribution used in CASMO-2E and the gap conductance used in RETRAN-02 and VIPRE-01. MICBURN-E provides the gadolinia cross sections used in CASMO-2E. CASMO-2E is used to perform the lattice physics analysis to generate the cross sections for SIMULATE-E and the inverse neutron velocity and total effective delayed neutron yield for SIMTRAN-E. NORGE-B is used to transfer the CASMO-2E data to SIMULATE-E and SIMTRAN-E. SIMULATE-E develops the three-dimensional macroscopic cross section data to be processed by SIMTRAN-E. SIMTRAN-E collapses the three-dimensional cross section data to one dimension and transfers the other nuclear parameters to RETRAN-02. STRODE is used adjust the moderator density feedback behavior and delayed neutron fraction data for input to RETRAN-02. RETRAN-02 is used to perform the transient analysis. VIPRE-01 is used to determine the ΔCPR during the transient based on the local peaking factors provided by FICE. STARS, if required, is used to perform the statistical assessment to demonstrate compliance with the fuel cladding integrity or PAFF limits.

5.3.3.4 Licensing Analysis Procedure

In the analysis of the generator load rejection without bypass, the following analysis assumptions are applied:

(a) The scram times are based on plant surveillance data. A statistical analysis is performed for this event, scram time statistics are included in development of the response surface as described in Appendix A.

(b) Scram initiation time delay is the maximum technical specification value.

(c) RPT time delay is the maximum technical specification value. An analysis is also performed without RPT to establish the appropriate operating limit for plant operation with RPT out of service.

(d) Relief valve opening setpoints are consistent with technical specifications.

(e) The analysis is performed at end of cycle conditions, with all control rods fully withdrawn.

(f) The analysis is performed at the most limiting point on the power/flow operating map, consistent with the license basis assumption on maximum power level.

(g) Feedwater temperature is determined by the RETRAN code, consistent with the system heat balance at the licensing power level. If the plant is allowed to operate with final feedwater temperature reduction to extend the operating cycle, the limiting feedwater temperature is calculated using a consistent set of nuclear input parameters.

(h) The turbine control valves operate in the full arc mode (all control valves are at the same position) and have a full stroke closure time, from fully open to fully closed, of 0.07 seconds. The closure time from their normal operational position is assumed to be proportional to the full stroke time. In actuality the valves are operated in the partial arc mode. However for this analysis, full arc mode is assumed which is conservative.

System actuations caused by low reactor water level trip setpoints, including main steam line isolation valve closure and high pressure makeup initiation, are not included in the simulation. These trips, should they occur, will be after the time of challenge or nearest approach to the event acceptance limits.

5.3.3.5 Sensitivity Studies/Justification of Procedure

RETRAN-02 and VIPRE-01 analyses were performed to determine the sensitivity of the calculated results to changes in input assumptions. The parameters considered in the sensitivity analysis were:

(a) Nuclear Model Parameters

Void reactivity
Doppler reactivity

Prompt moderator heating
Scram reactivity
Scram speed

(b) Core Thermal Hydraulics Parameters

Correlation parameters
Initial core flow
Core pressure drop
Core bypass flow fraction
Fuel pin nodalization
Initial core power

(c) Recirculation System Parameters

Recirculation loop inertia
Recirculation pump head
Jet pump inertia
Steam separator inertia
Jet pump pressure drop
Prompt RPT

(d) Steam Line Model Parameters

Steam line inertia
Steam line pressure drop

(e) Vessel and Loop Geometry Parameters

Vessel steam dome volume
Steam line volume
Steam line nodalization
Reactor core nodalization

There are several objectives for sensitivity studies, including:

- support verification of the licensing model;
- demonstrate conservatism in the licensing model; and
- quantify the impact of uncertainties in model parameters, input parameters, and modeling options.

RCPR, defined as ΔCPR/ICPR (where ICPR is the initial CPR), is used as the figure of merit in the sensitivity studies. The model parameter values used in the sensitivity studies and the calculated changes in RCPR are shown in Table 5.3.3-2. All parameters, except those marked with an asterisk, are included in the model uncertainty evaluation and are discussed in Appendix A.

The base case is performed with the highly conservative technical specification scram times. When normal scram time is used, the calculated RCPR is reduced by 0.062, the largest change found in the sensitivity study. Therefore, scram time is used as a response surface independent variable in the SCU analysis (See Appendix A). The steam line and reactor core nodalization cases demonstrate that the base model is conservative. The RPT inoperable case establishes the limits for plant operation with RPT out of service. Therefore, these parameters are not considered in the model uncertainty evaluation.

5.3.3.6 Typical Results

The base case analyzed is for technical specification scram time, rated core flow and with RPT operable. Plots of core power in terms of percent of nuclear boiler rated (%NBR), core average heat flux, core inlet flow, reactor vessel steam dome pressure, vessel steam flow, reactor vessel water level, and feedwater flow are provided on Figures 5.3.3-1 through 5.3.3-7.

A comparison of the Supply System and fuel supplier (ANF) calculated results for the increased core flow point for WNP-2 Cycle 4 are provided in Table 5.3.3-3. The Supply System calculated RCPR compares well with the ANF results. For the case with conservative technical specification scram time at 106% core flow with RPT, the Supply System calculates RCPR of 0.235 compared to 0.231 calculated by ANF. For the case with technical specification scram time at 106% core flow without RPT, the Supply System calculates RCPR of 0.277 compared to 0.259 calculated by ANF. Supply System results for the rated core flow point are shown in Table 5.3.3-4.

Table 5.3.3-1

Sequence of Events for Generator Load Rejection Without Bypass

Time(SEC)*	Events
0.0	Generator load rejection initiates fast closure of the turbine control valves.
0.0	Fast turbine control valve closure initiates signals for reactor scram, bypass valve operation, and RPT.
0.0	Turbine bypass valves fail to open.
0.07	Turbine control valves are fully closed.
0.19	Recirculation pump motor circuit breakers open causing decrease in core flow.
0.28	Control rods insertion starts.
1.36	Group 1 relief valves open to mitigate system pressure increase.
1.63	Group 5 relief valves open to mitigate system pressure increase.
**	Relief valves cycle closed/open to maintain system pressure.
**	Reactor vessel water level reaches L8 setpoint and feedwater pumps are tripped.
**	Low level (L2) reactor vessel water level is reached.
**	High pressure coolant inventory makeup systems are initiated.

* RETRAN simulation results.

** These events are beyond the RETRAN simulation time.

Table 5.3.3-2
Results of Generator Load Rejection Without Bypass Sensitivity Studies

	ΔRCPR	
Nuclear Model Parameters		
Void Coefficient (13%)	+0.018	
Doppler (-10%)	+0.005	
Prompt Moderator Heating (-25%)	+0.013	
Scram Reactivity (-10%)	+0.004	
Scram Speed (normal scram time)	-0.045	(1)*+
Core Thermal Hydraulics Parameters		
Code Correlation (kappa1±0.20)	+0.001	
Code Correlation (CGL±30%)	+0.003	
Code Correlation (CDB±20%)	+0.001	
Code Correlation (CHN±20%)	+0.001	
Initial Core Flow at 106%	+0.014	(2)*
Initial Core Flow at 106%, no RPT	+0.056	(2) (3)*
Core Pressure Loss Coefficients (-20%)	-0.002	
Initial Core Bypass Flow (-20%)	+0.003	
Fuel Pin Radial Nodes (+50%)	+0.004	
Core Power (+4%)	+0.003	
Recirculation System Parameters		
Recirculation Loop Inertia (+100%)	+0.007	
Recirculation Pump Head (-10%)	+0.003	
Jet Pump Inertia (+100%)	+0.008	
Separator Liquid Outlet Inertia (100%)	+0.001	
Separator Inlet Inertia (-30%)	+0.003	
Jet Pump Loss Coefficient (-20%)	+0.004	
No RPT	+0.040	(3)*
Steam Line Model Parameters		
Steam Line Inertia (+7%)	+0.007	
Pressure Loss Coefficient (-20%)	+0.007	
Vessel and Loop Geometry Parameters		
Vessel Dome Volume (-5%)	+0.005	
Steam Line Volume (-5%)	-0.002	
Steam Line Noding (7 → 13)	-0.003	(4)*
Active Core Noding (12 → 24)	-0.011	(4)*

* Not considered in model uncertainty evaluation.
(1) Scram time is a response surface independent variable.
(2) This is an analysis of the bounding core flow to establish the limit for operation at 106% core flow.
(3) This case establishes the limits for plant operation with RPT out of service.
(4) These cases demonstrate conservatism of the base model steam line and core nodalization.

\+ Normal scram time is the rod insertion time based on WNP-2 measured values. The normal scram times are cited in the

Table 5.3.3-2 (continued)

plant technical specifications and are approximately 2 standard deviations above the mean measured scram times. All other cases were performed with the technical specification scram speed which is a very conservative upper bound. A discussion of WNP-2 scram time is provided in Appendix A.

Table 5.3.3-3
Comparison of Supply System Results to ANF Results for Generator Load Rejection Without Bypass
---- Increased Core Flow Point ---

Technical Specification Scram Time, 106% Core Flow, RPT

	Supply System Result	Fuel Vendor Result
Maximum Neutron Flux (% Rated)	443	442
Maximum Core Average Heat Flux (% Rated)	134	125
RCPR	0.235	0.231
ΔCPR	0.30	0.30

Technical Specification Scram Time, 106% Core Flow, no RPT

	Supply System Result	Fuel Vendor Result
Maximum Neutron Flux (% Rated)	609	574
Maximum Core Average Heat Flux (% Rated)	145	131
RCPR	0.277	0.259
ΔCPR	0.38	0.35

Table 5.3.3-4
Supply System Results for
Generator Load Rejection Without Bypass
---- Rated Core Flow Point ---

Technical Specification Scram Time, Rated Core Flow, RPT

	Supply System Result
Maximum Neutron Flux (% Rated)	410
Maximum Core Average Heat Flux (% Rated)	132
RCPR	0.221
ΔCPR	0.28

Technical Specification Scram Time, Rated Core Flow, No RPT

	Supply System Result
Maximum Neutron Flux (% Rated)	558
Maximum Core Average Heat Flux (% Rated)	143
RCPR	0.267
ΔCPR	0.36

NOTE: Fuel vendor results are not available for these conditions.

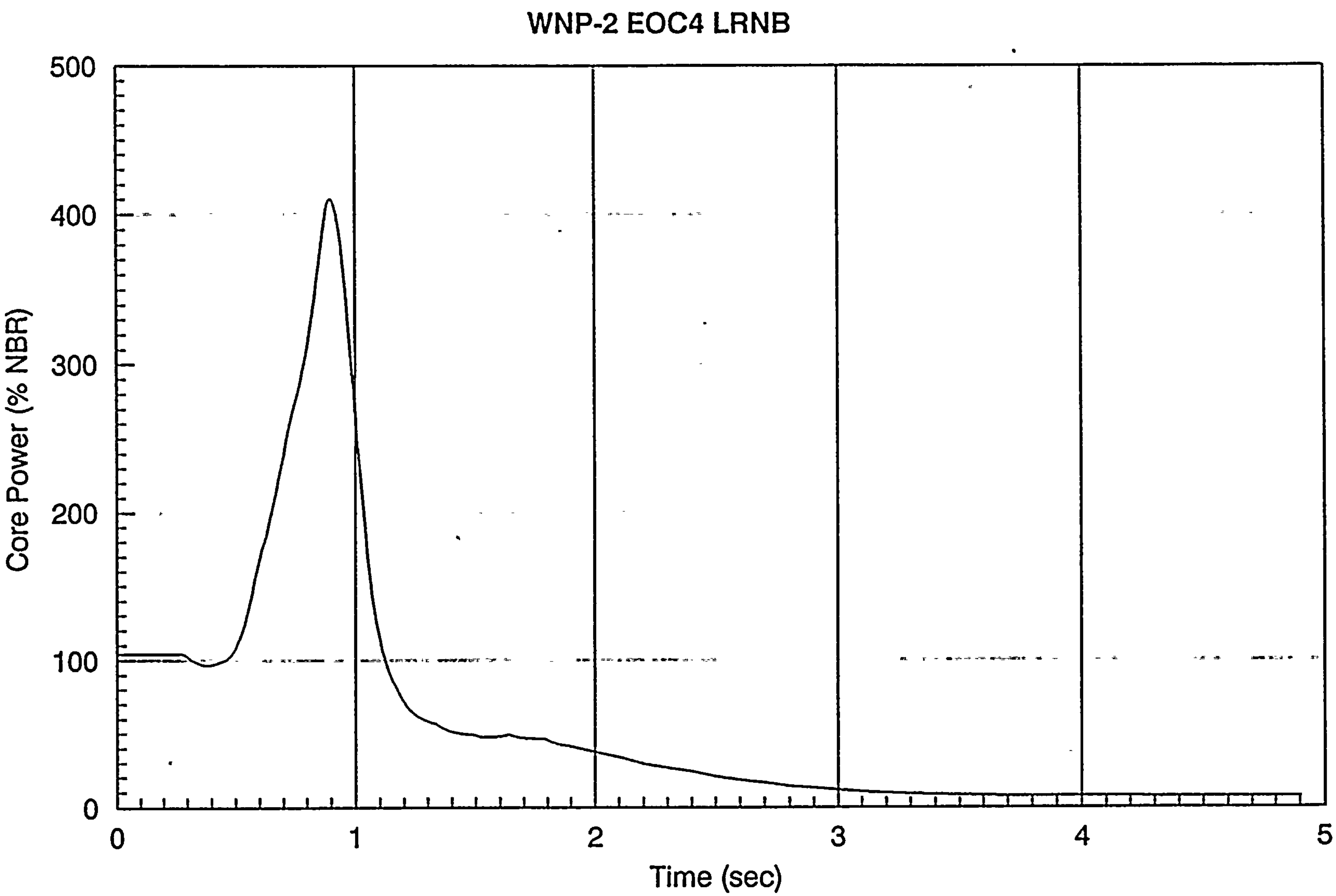

Figure 5.3.3-1 LRNB Results, RPT Operable, Tech. Spec. Scram Time

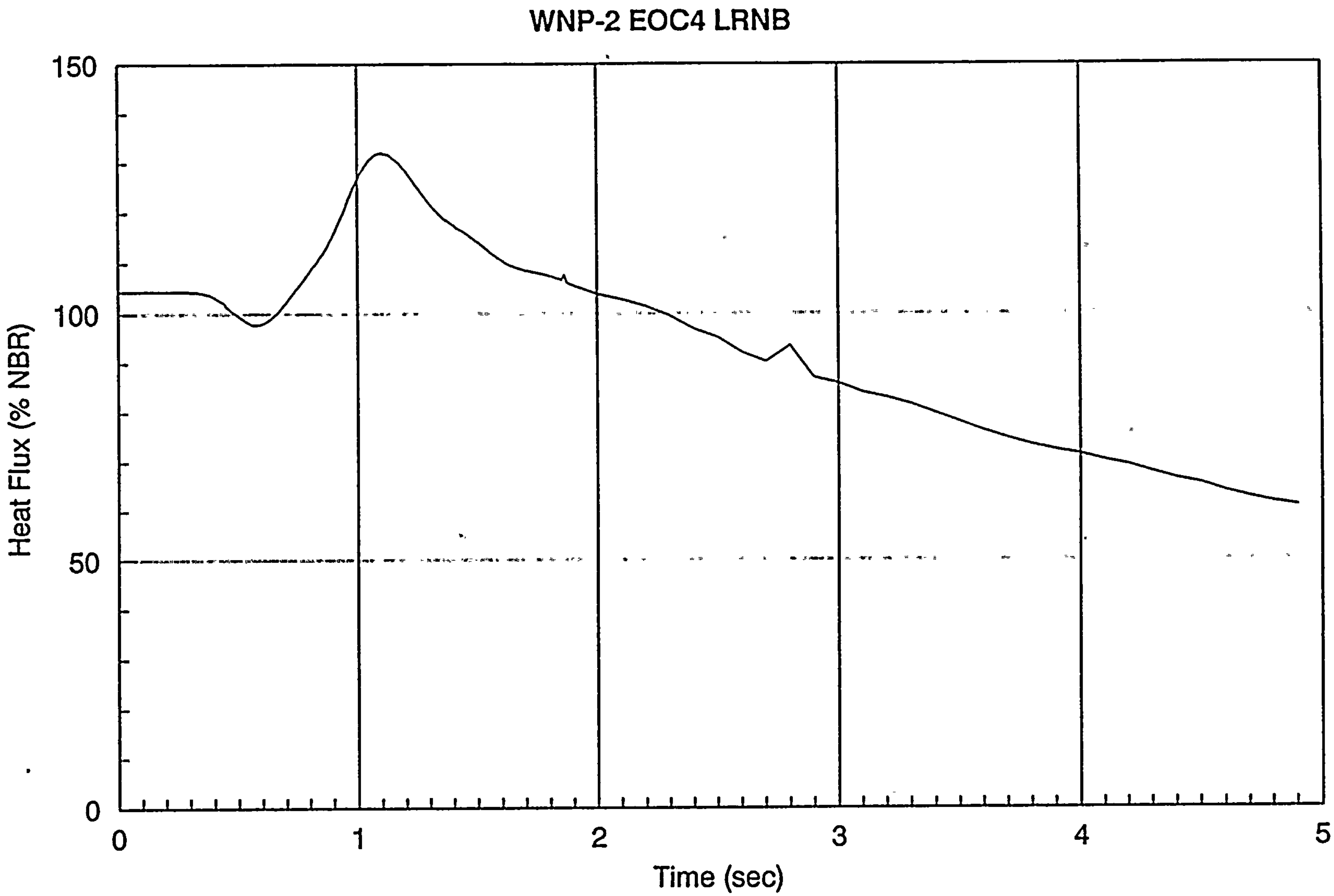

Figure 5.3.3-2 LRNB Results, RPT Operable, Tech. Spec. Scram Time

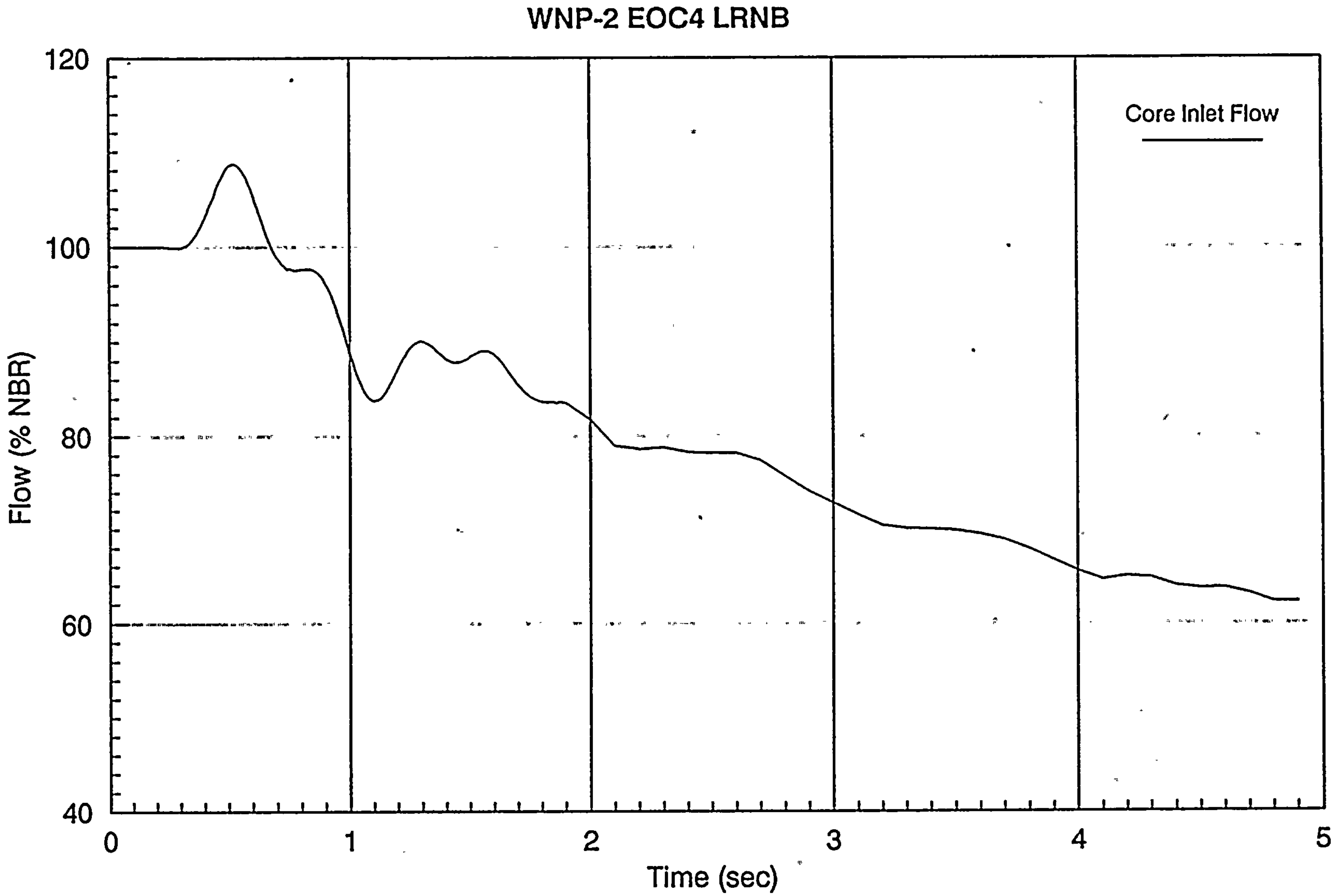

Figure 5.3.3-3 LRNB Results, RPT Operable, Tech. Spec. Scram Time

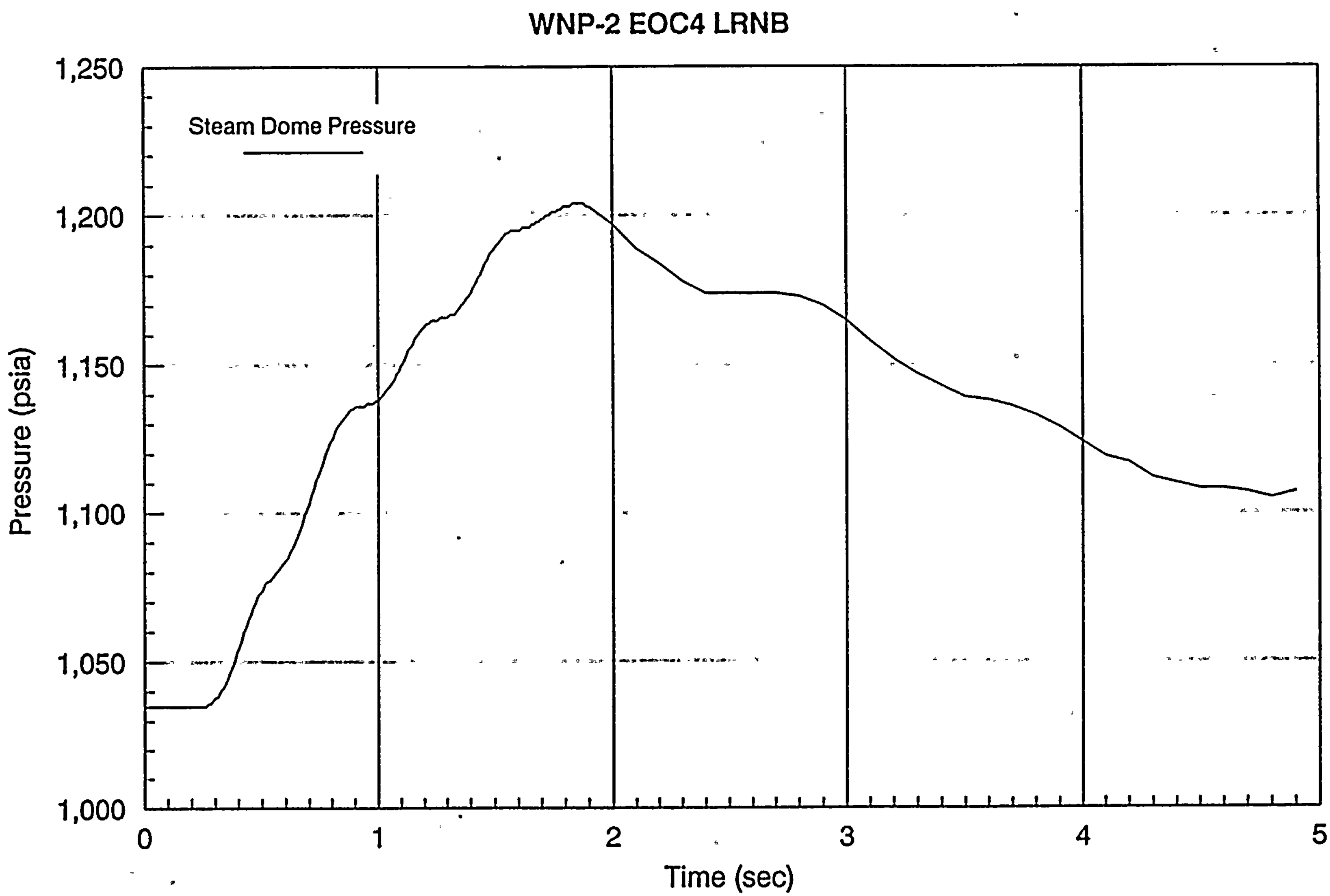

Figure 5.3.3-4 LRNB Results, RPT Operable, Tech. Spec. Scram Time

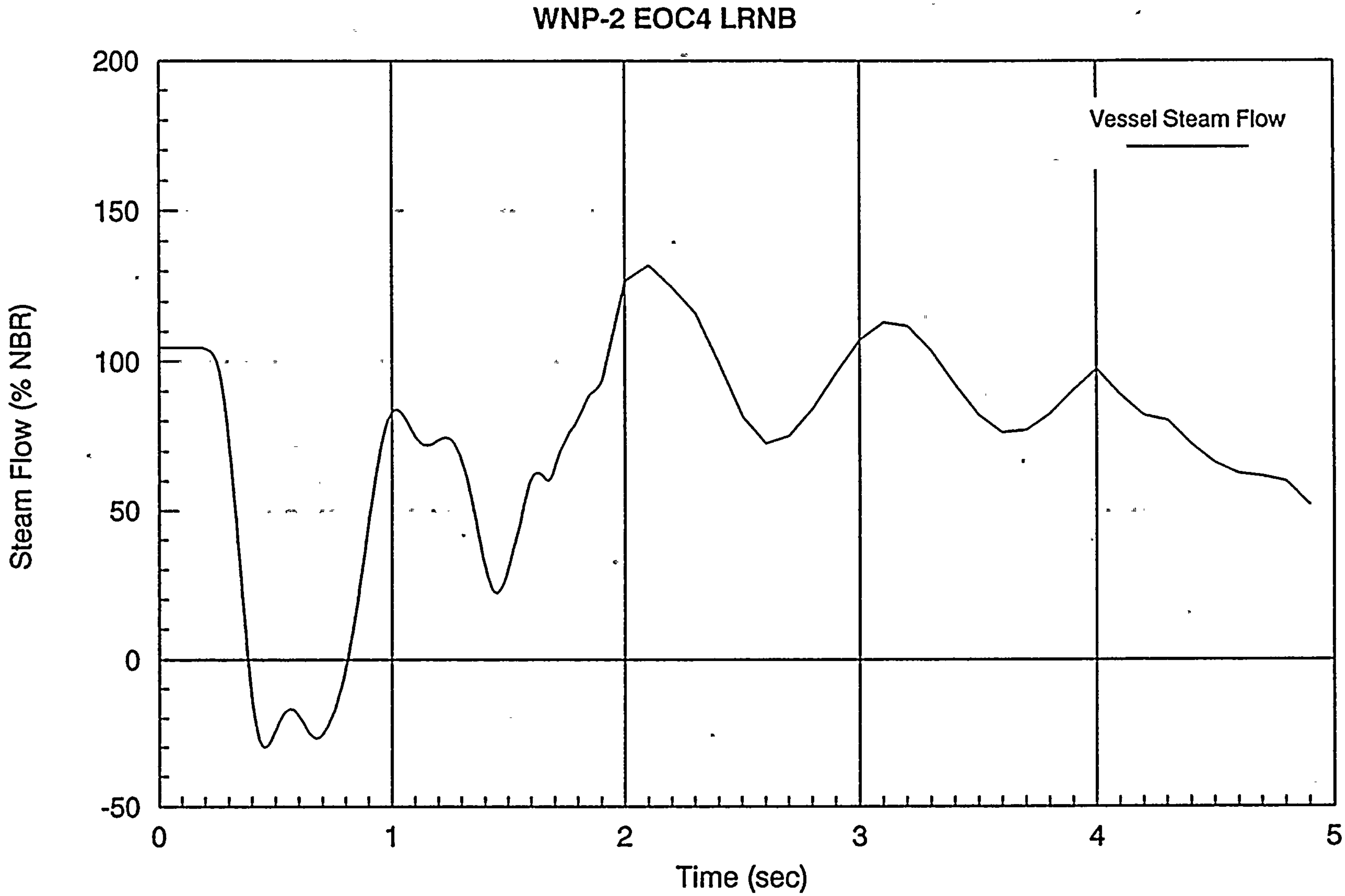

Figure 5.3.3-5 LRNB Results, RPT Operable, Tech. Spec. Scram Time

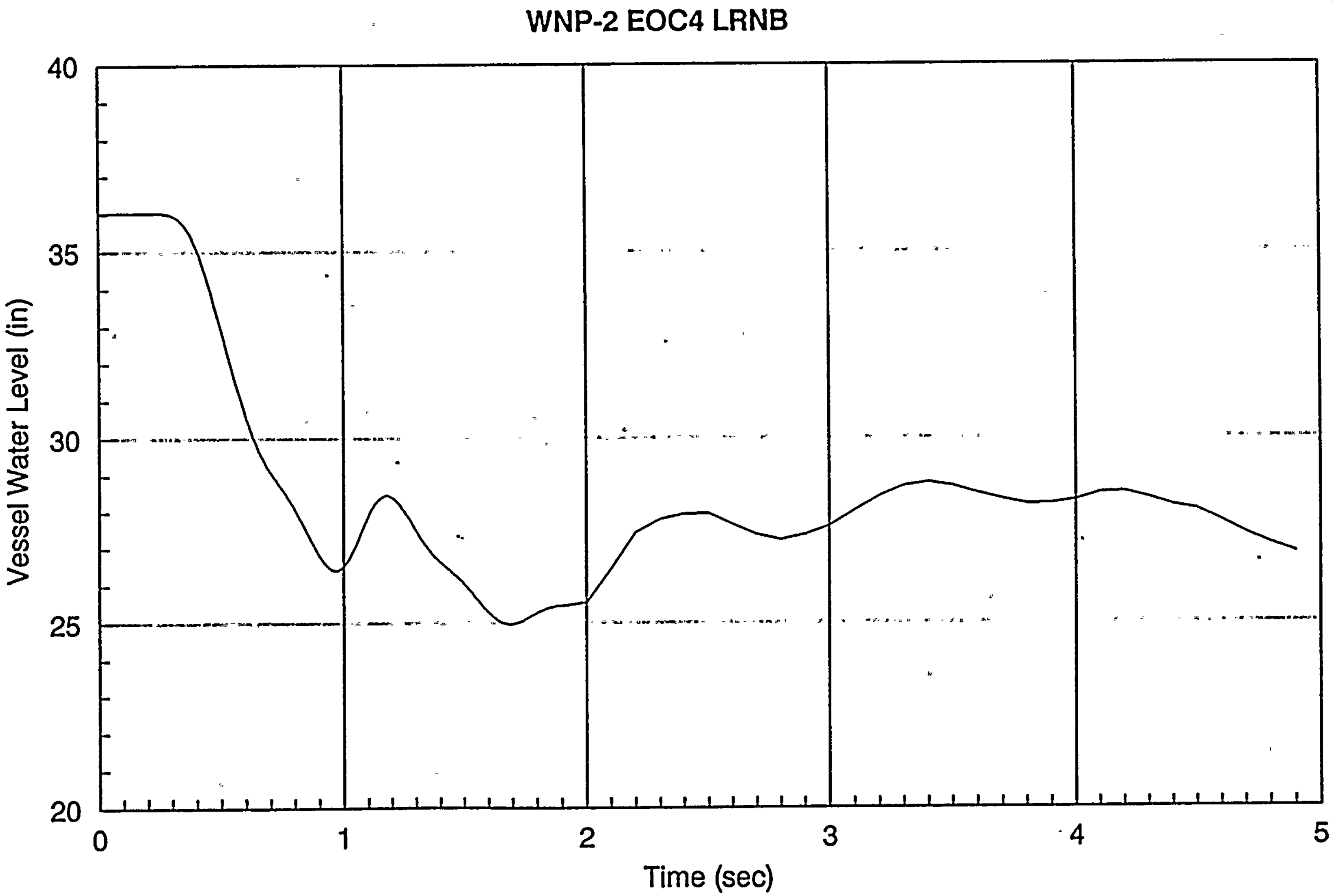

Figure 5.3.3-6 LRNB Results, RPT Operable, Tech. Spec. Scram Time

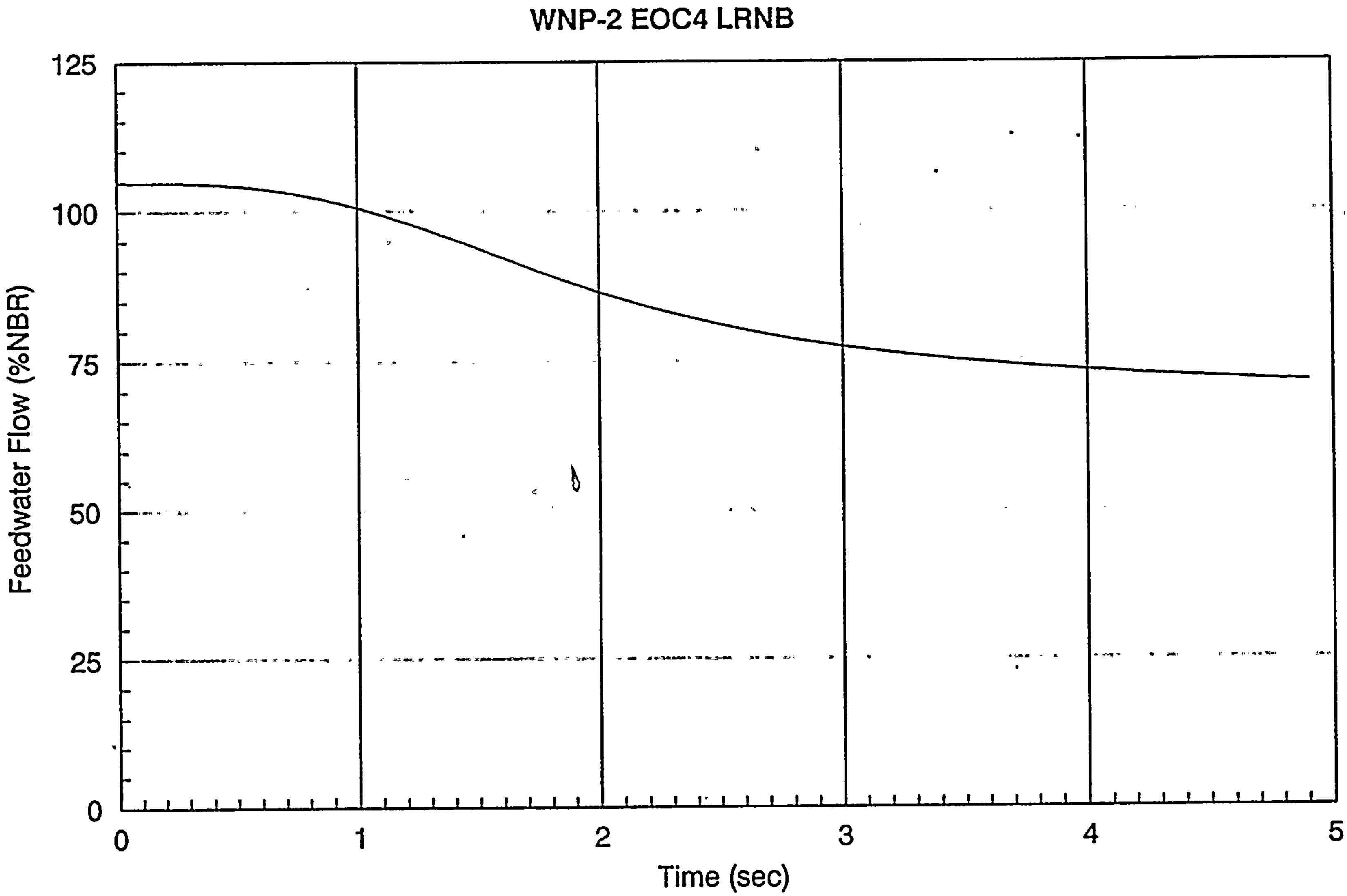

Figure 5.3.3-7 LRNB Results, RPT Operable, Tech. Spec. Scram Time

5.3.4 Control Rod Withdrawal Error at Power

5.3.4.1 Event Description

The control rod withdrawal error is initiated by an operator erroneously selecting and continuously withdrawing a control rod. Due to the positive reactivity insertion, both the core average power and the local power in the vicinity of the erroneously withdrawn control rod increase. The core average and local power increase continues until the RBM acts to inhibit further withdrawal, or the control rod reaches its fully withdrawn position. The turbine control valves will open to compensate for the increased steam flow until a new steady-state operating condition is reached.

5.3.4.1.1 Initial Conditions and Operational Assumptions

The number of possible control rod withdrawal error events is extremely large due to the number of control rods in the core and the wide range of exposures and power levels during an operating cycle. In order to encompass all of the possible control rod withdrawal errors which could conceivably occur, a limiting analysis is defined such that a conservative assessment of the consequences is provided. The conservative assumptions are:

(a) The postulated error is a continuous withdrawal of the rod which is expected to cause the maximum change in ΔCPR.

(b) The core is operating at rated conditions.

(c) The reactor is in its most reactive state and devoid of all xenon. This insures that the amount of excess reactivity which is controlled by the movable control rods is maximum.

(d) Furthermore, the operator has selected the control rod pattern in such a way as to approach thermal limits in the 36 fuel assemblies (6x6 array) in the vicinity of the control rod to be withdrawn. It should be emphasized that this control rod configuration is not consistent

with normal operating control rod patterns and is highly unlikely.

(e) The operator ignores all warnings during the transient, including RBM system alarms which must be reset in order to continue rod withdrawal.

(f) Failures have occurred in the local power range monitor (LPRM) strings that provide input to the RBM system (i.e., the four LPRM strings nearest to the control rod being withdrawn). Three failure combinations are included in the rod withdrawal error analysis: (1) the limiting single LPRM instrument, (2) the limiting LPRM string, and (3) the limiting set of two LPRM strings.

(g) One of the two RBM instrument channels is assumed to be bypassed and out of service. The A and C elevation LPRM chambers input to one channel while the B and D elevation LPRM chambers input to the other. The channel with the greatest response is assumed to be bypassed.

These conservative assumptions provide a high degree of assurance that the transient, as analyzed, bounds all control rod withdrawal error events that could reasonably be expected to occur.

5.3.4.1.2 Operator Actions

As indicated above, the operator is assumed to have set up a very unlikely control rod pattern in which the "error" rod is that rod which could cause the most serious event consequences, and the remaining rod pattern enhances the response to withdrawal of the error rod. The operator proceeds to withdraw the error rod until it is blocked by the RBM or is fully withdrawn. The operator fails to stop the withdrawal on either high LPRM alarms or RBM alarms which require acknowledgement and should result in no further rod withdrawal. The licensing analysis is based on highly abnormal control rod patterns and operating conditions and assumes that the operator ignores all alarms and warnings and continues to withdraw the control rod.

5.3.4.1.3 Event Acceptance Limits

The acceptance limits for this event are a MCPR $\geq$ fuel cladding integrity limit and the LHGR $\leq$ PAFF limits. Compliance to the fuel cladding integrity limit is demonstrated by assuring that the operating limit MCPR is greater than or equal to the fuel cladding integrity limit MCPR (which assures that greater than 99.9% of the fuel rods in the core are not expected to experience boiling transition) plus the ΔCPR during the event. Compliance to the PAFF limits is demonstrated by assuring that the change in fuel design limit ratio (ΔFDLRX) is less than 0.2.

5.3.4.1.4 Analysis Considerations

This section describes the key analysis considerations applicable to the control rod withdrawal error at power event. It includes: (1) a description of the phenomena occurring during the event that have a significant impact on the event consequences; (2) a discussion of the system performance characteristics that can significantly affect the course of the event; and (3) a discussion of the performance characteristics of the important components as they relate to the event consequences.

5.3.4.2.1 Key Phenomena

The event begins with the continuous withdrawal of a control rod. This action results in a significant change in local power and power distribution and a relatively mild change in core average power and system conditions. Therefore, accurate simulation of the core phenomena is the key to the analysis of the control rod withdrawal error event.

The phenomena important in the core region are the reactivity effects that contribute to changes in the local neutron flux and the fuel energy generation for coolant heating. The primary reactivity effects are the reactivity insertion due to control rod withdrawal and the feedback effects of void and Doppler reactivity.

During the control rod withdrawal error event, the withdrawal of the control rod is responsible for the initial positive reactivity insertion. This causes an increase in local power and core average power level which increases fuel temperature and generates additional steam voids. The local and core average power increase continues until the control rod movement is terminated by the control rod reaching its full out position or by action of the RBM system. The increased steam voids and Doppler reactivity will limit the power increase and, upon termination of the control rod movement, a new equilibrium power level will be established.

The control rod withdrawal error is simulated with the use of a reactor physics code (lattice physics and three-dimensional simulator) by calculating the core response at a series of quasi-steady state points. The amount of control rod movement is dependent on the RBM signal and trip setpoints which is dependent on the measurement of neutron flux by the 4 LPRM strings surrounding the control rod being withdrawn. The analysis is generally performed at 6 inch increments which correspond to two control rod notch positions so that the location of the rod block position can be determined.

In evaluating the core phenomena, the reactor is initialized at full power with zero xenon condition with the control rods adjusted to have a critical core condition. The error rod is fully inserted.

In the vicinity of the error rod, at least one fuel assembly is close to the core wide MCPR and one fuel assembly is close to the LHGR operating limits prior to the control rod withdrawal. During the quasi-steady state evaluations, the critical eigenvalue is maintained and core and local power are allowed to change due to the reactivity insertion from the control rod being withdrawn. The xenon concentration is maintained at its zero condition. Because this is a localized reactivity insertion event, the 36 fuel assemblies surrounding the error rod are evaluated to establish the

limiting fuel assemblies from both an MCPR and LHGR standpoint during the event.

The quasi-steady state analyses are used to calculate changes in MCPR and maximum LHGR during the event. The results of the analysis are then used to demonstrate a compliance to the SAFDLs (fuel cladding integrity and PAFF limits). Because this is a highly localized transient, fuel densification is considered in the evaluation of the maximum LHGR. A 2.2% power spiking penalty is applied to the control rod withdrawal error maximum LHGR to account for the fuel densification effect.

The evaluation of this event employs the use of the three-dimensional simulator code, and the external system effects are treated as code inputs for the quasi-steady state evaluations. Use of the quasi-steady-state approach is justified on the basis of the slow reactivity insertion rate. The core has sufficient time to equilibrate (both neutron flux and heat flux are in phase). Conservatism in the event conditions is sufficient so that code inputs that reflect external system operation can be simplified.

The core power increase results in a slight increase in steam flow which slightly increases reactor pressure and core pressure drop. Feedwater flow increases to maintain reactor water level, considering the increased steam flow. In the manual flow control mode, recirculation pump speed is held constant; therefore, the increase in core pressure drop will result in a slight decrease in core flow. These effects are considered to be second order and do not contribute significantly to the analysis of the control rod withdrawal error.

5.3.4.2.2 Systems Considerations

The control rod withdrawal error event is initiated by an operator error in selecting and withdrawing a control rod from an abnormal control rod pattern. All systems are assumed to operate as designed and continue to function throughout the event. The RBM

system is assumed to initiate at its pre-selected setpoints. The RBM failures assumed in the analysis are described in section 5.3.4.1.1(f) and (g).

5.3.4.2.3 Component Performance Characteristics

The RBM system minimizes the consequences of a control rod withdrawal error by blocking the motion of the control rod before the safety limits are exceeded.

The RBM has three trip levels that remove the rod withdrawal permission. The trip levels may be adjusted and are nominally separated in increments of 8% of reactor power. The highest trip level is set so that the fuel cladding integrity safety limit is not exceeded. The lower two trip levels are intended to provide a warning to the operator. Typical settings are 106%, 98%, and 90% of initial steady-state operating power at 100% core flow. The highest level trip is controlled by a technical specification. The trip levels are automatically varied with reactor coolant flow for protection at lower flows. The variation is set to assure that event acceptance limits are not exceeded at the indicated coolant flow. The operator may encounter up to three trip points depending on the core power at the start of a given control rod withdrawal. The two lower points may be reset by manual operation of a push button. The reset permissive is actuated (and indicated by a light on the RBM panel) when the RBM reaches 2% power less than the trip setpoint. The operator should then assess the local power and either reset or select a new rod for withdrawal. The highest power trip setpoint may not be reset during control rod withdrawal operations.

The RBM processes LPRM signals from the four LPRM strings that surround the rod selected for withdrawal. There are two RBM channels; an A channel utilizing the 8 LPRM signals from the A and C elevations, and a B channel utilizing the 8 LPRM signals from the B and D elevations.

5.3.4.3 Methodology/Integration of Codes and Analysis

Analysis of the control rod withdrawal error primarily utilizes the steady-state SIMULATE-E three-dimensional simulator code for the calculation of core response to the withdrawn rod. Lattice physics input to SIMULATE-E is through the ESCORE, MICBURN-E, CASMO-E, and NORGE-B path. TLIM reads the SIMULATE-E restart file and calculates MCPR and maximum LHGR as a function of the notch position of the withdrawn rod. CALTIP reads the SIMULATE-E restart file and provides a calculation of the response of every LPRM in the core. The RBLOCK code calculates the RBM output signal based on the LPRM signals that are calculated by CALTIP. RBLOCK calculates the MCPR and the maximum LHGR at the position of the rod block for a large number of LPRM failure combinations. The maximum change in these fuel parameters is utilized in determining the adequacy of the operating limits relative to the event acceptance limits. (See Section 2 and Figure 2-1 for an overview of the overall WNP-2 reactor analysis methodology computer code sequence.)

5.3.4.4 Licensing Analysis Procedure

This section describes the licensing analysis procedure. Section 5.3.4.5 provides a discussion of sensitivity analyses and other factors that justify the appropriateness of this procedure.

(a) The WNP-2 reactor core for the cycle of interest is modeled with SIMULATE-E at rated power and flow.

(b) The exposure point in the cycle is found which has maximum hot excess reactivity with all rods out.

(c) The xenon concentration is set to zero.

(d) An error rod is selected based on the maximum average power level of the four bundles which are associated with that rod.

(e) A control rod pattern is developed to achieve criticality which causes at least one fuel assembly in the vicinity of the error rod (within a 6 by 6 fuel assembly array) to be near MCPR and one fuel assembly to be near maximum LHGR operating limits. In some cases, it may be the same

fuel assembly that satisfies these criteria. The error rod is fully inserted.

(f) Quasi-steady state calculations are made at several control rod positions as the control rod is stepped out of the core from the fully inserted position, and the power is adjusted to achieve criticality.

(g) The response of each RBM channel is calculated for each control rod position. Calculations include results with failure of any single LPRM string, any pair of LPRM strings, and all combinations of a single LPRM failure in each of the two levels of the RBM channels.

(h) The limiting MCPR and maximum LHGR are determined for the worst mode of LPRM failures and the assumption that the most responsive RBM channel (typically channel B) is inoperative. The limiting values are associated with the rod position where the RBM blocks the rod at the technical specification trip setpoint (typically 1.06) or at the fully withdrawn position if a block does not occur.

5.3.4.5 Sensitivity Studies/Justification of Procedure

The following sensitivity studies were performed to determine the sensitivity of the control rod withdrawal error analysis to various parameters and provide justification for the licensing analysis procedure: (1) effects of control rod pattern, which led to the selection of the limiting control rod pattern; (2) the effect of xenon inventory, which led to the selection of zero xenon state; (3) the effect of time in cycle exposure and core reactivity, which led to the selection of most reactive state; (4) the effects of core power and flow, which led to the selection of the rated power and flow case; (5) the effects of LPRM and RBM failure modes; which led to the selection of limiting cases to be considered; (6) effects of error rod location, which led to the selection of rod location for the position of maximum four bundle average power; and (7) the effects of core pressure and inlet subcooling increasing during the event. The results of these sensitivity studies are described in more detail below.

As a result of the sensitivity studies, it was determined that the most sensitive parameter with respect to the control rod withdrawal

error analysis is the selection of the control rod pattern. In the licensing analysis procedure, it is required that at least one fuel assembly in the 6x6 fuel assembly array surrounding the error rod be on or near estimated MCPR limits, and one fuel assembly in the 6x6 fuel assembly array surrounding the error rod be on or near LHGR limits. Control rod patterns that result in an increase the power in the vicinity of the control rod to be erroneously withdrawn create large control rod worths and increase the event consequences. This results because the neutronic coupling is increased as the limiting fuel assemblies are moved closer to the control rod to be withdrawn. In addition, the core is required to be in a xenon free condition while maintaining criticality. The xenon free assumption results in the most reactive core conditions and largest control rod inventory, which leads to a more conservative power distribution in the vicinity of the fully inserted control rod.

The requirements associated with the selection of the limiting control rod pattern lead to unrealistic control rod patterns which are not expected to be encountered during normal operation, because of the excessive power peaking that would result. To demonstrate the conservatism of the selection of the limiting control rod pattern, two different control rod patterns were evaluated: (1) using the licensing analysis procedure and (2) using a conservative control rod pattern based on equilibrium xenon conditions. For the first case, complete withdrawal of the control rod led to a ΔCPR of 0.23. For the second case, complete withdrawal of the control rod led to a ΔCPR of 0.15. In the second case, the control rod pattern was developed based on the expected control rod patterns anticipated during normal operation based on an equilibrium xenon condition; however, they were made conservative by having a fully inserted control rod that could be erroneously withdrawn and adjusting the remaining control rods to increase the fuel assembly powers in the vicinity of the fully inserted control rod.

The xenon inventory assumption has a very strong influence on the development of the control rod pattern because of the relatively

large amount of negative reactivity associated with equilibrium xenon conditions. Once the control rod pattern has been established, the effect of xenon has a relatively insignificant effect. The same rod patterns were used in the evaluation of the control rod withdrawal error for both the xenon free and equilibrium xenon conditions, which ignores the criticality requirements. For these two cases, a difference in ΔCPR of less than 0.01 was seen over the entire range of control rod withdrawal. Therefore, the selection of the xenon free condition for analysis is conservative because the rod pattern developed with a xenon free condition gives about 0.08 higher ΔCPR than the rod pattern developed with an equilibrium xenon condition. This shows that the xenon condition is only important in establishing the control rod pattern.

Cycle exposure that has the greatest hot excess reactivity is used in the control rod withdrawal error analysis because it is associated with the greatest fuel assembly power with all control rods out and the largest control rod inventory during the operating cycle, and therefore, it has the greatest amount of flexibility in establishing the control rod pattern. To determine the effect of the most reactive state assumption, sensitivity studies were performed that demonstrated that exposure has a relatively large effect on the event consequences. Two cases were run using the same control rod patterns, for two different exposures: (1) at 0.2 GWD/MT, most reactive exposure and (2) 2 GWD/MT. For these cases, the ΔCPR was within 0.10 for all control rod withdrawal positions, with the most reactive core exposure being more limiting.

Core power and flow conditions were investigated to determine the sensitivity of the initial core operating state to the consequences of the control rod withdrawal error event. Four separate cases were analyzed to determine the effects of the operating state on the event consequences: (1) rated power and core flow; (2) 90% power and 87% rated core flow; (3) 90% power and 100% rated core flow; and (4) 105% steam flow (104.2% power) and rated core flow.

All of the cases were run with the same control rod pattern except for the 90% power and 100% core flow case. For the 90% power and 100% core flow case, control rods were inserted to satisfy the criticality constraints.

As the core flow is reduced along the 100% rod line, the RBM setpoint at which the control rod block will be initiated is decreased, because of the flow referencing circuitry. For example, a rod block setpoint of 106% at rated core flow will have a rod block setpoint of about 98% at 87% core flow. Comparison of the control rod withdrawal error analysis at the 90% power and 87% core flow case with the rated power and flow case, results in an increase in 0.002 ΔCPR. Therefore, the change along a constant rod line is insignificant.

As the core flow is increased, the RBM setpoint at which the control rod block will be initiated is effectively increased. For a constant operating power, the increased core flow case will have a higher ΔCPR. This phenomenon is demonstrated by comparison of the 90% power/87% flow case (100% rod line) with the 90% power/100% flow case (90% rod line). The ΔCPR for the higher flow case is 0.09 greater than the low flow case. However, the operating MCPR for the higher flow case is 0.12 greater than the low flow case. Therefore, the increase in MCPR operating margin due to the higher flow more than compensates for the increase in ΔCPR due to the higher RBM setpoint. Consequently, the use of high rod line is appropriate.

As the power level is reduced while operating at a constant core flow, a control rod withdrawn in error can be withdrawn further before a control rod block will occur. As a result, the control rod withdrawal error for a reduced power level at the same core flow will have higher ΔCPR. This phenomenon is demonstrated by comparison of the rated power and flow case with the 90% power and 100% flow case. The ΔCPR for the lower power and flow case is about 0.09 greater than for the rated power and flow case. However, the operating MCPR for reduced power case is 0.20 greater

for the 90% power and rated core flow case than the rated power and flow case. Therefore, the operating MCPR increases faster than the change in ΔCPR, and the use of higher power is conservative.

Comparison of the rated power and flow case to the 104.2% power case shows the change in ΔCPR for all RBM setpoints is less than ±0.01. The rated power and flow case is used in the licensing analysis process because it provides additional control rod inventory that can be used in the selection of the limiting control rod pattern. Therefore, the above sensitivity studies shows that the use of rated power and flow case for licensing is adequate.

The assumed availability of specific LPRM inputs to the RBM system directly effects the results of the analysis of the control rod withdrawal error. There is a wide variability in the RBM response for different failure potential combinations, and no single failure combination within the constraints of the system design can be demonstrated to be the most limiting for all combinations of control rod patterns and potential LPRM failure combinations. This results because of differences in the RBM response to different control rod errors and a different sensitivity to the axial location of the control rod being withdrawn. To assure a conservative assessment of the possible LPRM failure combinations, the response of each RBM channel is calculated for each control rod position. Calculations include results with failure of any single LPRM string, any pair of LPRM strings, and all combinations of a single LPRM failure in each of the two levels of the RBM channels. The limiting MCPR and LHGR are determined for the worst mode of LPRM failures and the assumption that the most responsive RBM channel (typically channel B) is inoperative. The limiting values are associated with the rod position where the RBM blocks the rod at the technical specification trip setpoint (typically 1.06) or at the fully withdrawn position if a block does not occur. This procedure assures that there is only a very low probability that a more limiting failure combination could occur.

In the licensing analysis procedure, the error rod location is selected based on the maximum average power level of the four bundles which are associated with that rod. The maximum average power level of the four bundle array is selected to assure that a very reactive core cell is used for the location of the control rod to be erroneously withdrawn. To assess the conservatism of this assumption, the second highest average power level of the four bundle array in the same sequence (Sequence A) was analyzed consistent with the remaining license analysis procedure requirements. The change in ΔCPR for the second highest average power level of the four bundle array was 0.07 less than the maximum power level of the four bundle array near the expected RBM setpoint. Therefore, the highest average power level of the four bundle array is used.

As the control rod error progresses, there is a small increase in reactor pressure and inlet subcooling. Analyses were performed both with the assumption of constant pressure and inlet subcooling, and with the calculated change in pressure and inlet subcooling during the event. It was determined that there was a difference of less than 0.01 change in ΔCPR in these cases. Therefore, the assumption of constant pressure and inlet subcooling is used in the license analysis procedure.

5.3.4.6 Typical Results

The licensing analysis procedure for evaluating the control rod withdrawal error event is described in Section 5.3.4.4. This procedure has been used to analyze the control rod withdrawal error for Cycle 4. Table 5.3.4-1 provides the input parameters and initial conditions used in the evaluation of the control rod withdrawal error for Cycle 4. The specific control rod pattern used in the analysis is shown on Figure 5.3.4-1. The analysis results are provided in Table 5.3.4-2. A comparison of the results of WNP-2 reactor analysis methodology to the fuel supplier (ANF) for Cycle 4 is shown on Table 5.3.4-3. Therefore, it is concluded that the Supply System licensing analysis procedure is reasonable

yet conservative based on the evaluation of the analysis conservatisms and uncertainties and the comparison with the ANF results.

Table 5.3.4-1

Input Parameters and Initial Conditions
for the Control Rod Withdrawal Error

Thermal Power (MWt)	3323
Cycle 4 Exposure (GWD/MT)	0.2
Core Average Exposure at 0.2 GWD/MT (GWD/MT)	11.35
Xenon Concentration	None
Average Initial LHGR (kW/ft)	5.44
Maximum Initial LHGR (kW/ft)	14.44
MCPR	1.29*
Location of MCPR Assembly	35-30
Location of Maximum Worth Control Rod	30-31
Core Coolant Flow Rate (Mlbm/hr)	108.5
Reactor Coolant Pressure (psia)	1035
Core Coolant Inlet Enthalpy (Btu/lbm)	527.0
Control Rod Pattern	Figure 5.3.4-1
RBM Trip Setpoint	106

* Value used for analysis purposes

Table 5.3.4-2

**Results of the Control Rod Withdrawal Error Analysis
Supply System Methodology**

Rod Block Monitor Setpoint (%)	ΔCPR	ΔFDLRX
103	0.14	0.026
104	0.16	0.053
105	0.18	0.085
106	0.19	0.117
107	0.20	0.136
108	0.21	0.152
109	0.23	0.152

Table 5.3.4-3

**Results of the Control Rod Withdrawal Error Analysis
Supply System Methodology Compared to Fuel Supplier Methodology**

Rod Block Monitor Setpoint (%)	Supply System ΔCPR	Fuel Supplier ΔCPR
106	0.19	0.17
107	0.20	0.18
108	0.21	0.20

	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58
59					-	-	-	-	-	-	-				
55				14	-	0	-	0	-	0	-	14			
51			-	-	-	-	-	-	-	-	-	-	-		
47		14	-	0	-	0	-	0	-	0	-	0	-	14	
43	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
39	-	0	-	0	-	12	-	32	-	12	-	0	-	0	-
35	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
31	-	0	-	0	-	32	-	0*	-	32	-	0	-	0	-
27	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
23	-	0	-	0	-	12	-	32	-	12	-	0	-	0	-
19	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
15		14	-	0	-	0	-	0	-	0	-	0	-	14	
11			-	-	-	-	-	-	-	-	-	-	-		
07				14	-	0	-	0	-	0	-	14			
03					-	-	-	-	-	-	-				

- Fully Withdrawn Control Rod

* Error Rod

Figure 5.3.4-1.
Control Rod Positions for the
Control Rod Withdrawal Error Analysis

5.3.5 Fuel Loading Error--Mislocated Fuel Assembly

5.3.5.1 Event Description

The mislocated fuel assembly fuel loading error is the postulated occurrence of the loading of a fuel assembly in an improper location. This causes a discrepancy between the design core configuration and the actual core configuration. It is assumed that the loading error is not detected during the core loading and verification process and that the plant operates throughout the cycle as constrained by the core operating limits for the design core configuration. Three errors must take place for this event to occur. First, a fuel assembly must be loaded into the wrong location. Then, the fuel assembly intended for that location must itself be placed in an improper location or not loaded in the core, creating an accounting discrepancy. Finally, the loading error must be overlooked during the final core configuration verification.

The consequences of the fuel loading error - mislocated fuel assembly are dependent upon the exposure and enrichment differences between the fuel assembly that has been incorrectly loaded and the fuel assembly that has been designed to be in the location. The event with the most severe consequences can occur at any exposure at which an assembly is at its peak reactivity and has been mislocated in an unmonitored location designed for a low reactivity assembly. It is then assumed that the operator has the capability for developing a control rod pattern that places fuel assemblies in the core on the core operating limits.

Because of the low probability of this event, it is considered an accident in the safety analysis process, and no other events or equipment failures are assumed to occur while the plant is operating with a fuel assembly loading error. However, the fuel cladding integrity limit has been adopted as the figure of merit for this event.

5.3.5.1.1 Initial Conditions and Operational Assumptions

The following plant operational conditions and assumptions form the principal bases for the analysis of the fuel loading error - mislocated fuel assembly event:

(a) A mislocated fuel assembly fuel loading error with the worst consequences has occurred and gone undetected.

(b) The plant is operating at rated conditions.

(c) The remaining NSSS operating parameters are consistent with normal plant operation.

(d) All of the plant control systems function normally.

5.3.5.1.2 Operator Actions

The operator is assumed to follow normal plant operating procedures because there is no information to indicate the existence of a fuel assembly loading error.

5.3.5.1.3 Event Acceptance Limits

The acceptance limit for this event is MCPR $\geq$ fuel cladding integrity safety limit. Compliance with this limit is demonstrated by assuring that the operating limit MCPR is greater than or equal to the MCPR which assures that greater than 99.9% of the fuel rods in the core are not expected to experience boiling transition plus the change in ΔCPR.

5.3.5.2 Analysis Considerations

This section describes the key analysis considerations applicable to the fuel loading error - mislocated fuel assembly event. It includes: (1) a description of the phenomena occurring during the event that have a significant impact on the event consequences; (2) a discussion of the system performance characteristics that can significantly affect the course of the event; and (3) a discussion

of the performance characteristics of the important components as they relate to the event consequences.

5.3.5.2.1 Key Phenomena

The fuel assembly mislocation can result in a significant change in the local power distribution. Two potentially significant situations exist: (1) misloading a fuel assembly into a monitored location designed for a higher reactivity fuel assembly; and (2) misloading a fuel assembly into an unmonitored location designed for a lower reactivity fuel assembly.

In the first case, the readings of the adjacent local power range monitors (LPRMs) decrease. Because there are no LPRMs in the symmetrical locations, the operator assumes that they have the same characteristics as the monitored locations. Should the operator select a control rod pattern that increases power based on the monitored LPRM readings, the assemblies in the corresponding symmetrical locations could exceed design limits.

In the second case, the power in the mislocated fuel assembly is inferred from the LPRMs in the correctly loaded symmetrical locations. Therefore, the mislocated fuel assembly is operating at higher power than assumed in the core design. The core operating limits for this assembly can be exceeded if the operator selects a control rod pattern that increases power in the monitored and symmetrical location to the core operating limits based on the assumed core configuration. From analyses performed by the Supply System, it has been shown that this will be the most limiting case because of the coupling effect of placing a highly reactive fuel assembly near another highly reactive fuel assembly.

5.3.5.2.2 Systems Considerations

All systems are assumed to be in normal power operational modes and continue to operate throughout the operating cycle.

5.3.5.2.3 Component Performance Characteristics

Because the mislocated fuel assembly can be in an unmonitored location, it is assumed that it will not be detected by the core monitoring system. Therefore, control rod patterns may be selected that would put the limiting fuel assemblies in the core on thermal limits. This causes the mislocated fuel assembly to exceed normal operating thermal limits.

5.3.5.3 Methodology/Integration of Codes and Analysis

The fuel loading error - mislocated fuel assembly utilizes the SIMULATE-E three-dimensional BWR simulator code as the primary analysis code. SIMULATE-E is used to calculate the reactor power, power distribution, fuel assembly flow rates for both the correctly loaded core and the core containing the mislocated fuel assembly. The lattice physics input to SIMULATE-E is provided by CASMO-2E through NORGE-B. The MICBURN-E code is used to determine the gadolinia cross sections used in CASMO-2E and ESCORE provides the fuel temperature distribution. TLIM, based on SIMULATE-E input, is used for the thermal limits evaluation. (See Section 2 and Figure 2-1 for an overview of the overall WNP-2 reactor analysis methodology computer code sequence.)

5.3.5.4 Licensing Analysis Procedure

The mislocated assembly analysis is performed on the basis of the final design depletion through the cycle with control rods present. Based on sensitivity analysis, it has been shown that the worst mislocated fuel assembly will occur when a high reactivity assembly is loaded in an unmonitored location. Also, based on sensitivity studies, it has been shown that the worst mislocated high reactivity fuel assembly results occur in the early part of a cycle with control rods present.

The selection of the high reactivity assembly is performed by selecting any assembly that may have a high reactivity at any time

in the cycle. Then a cycle depletion is performed using the same control rod patterns as the final design depletion for each assembly type that will lead in reactivity (at any time in the cycle) in its mislocated position. The selection of the mislocated position is chosen by two methods. The first method finds the highest four assembly average CPR in an unmonitored location based on based on each depletion step in the final design (note: this location varies throughout the cycle). The high reactivity assemblies are then mislocated into the highest CPR position within the highest four assembly average CPR. The second method is to find the lowest CPR unmonitored location at each depletion step in the final design, and the high reactivity assemblies are loaded face adjacent to the low CPR locations (note: this location varies throughout the cycle). The mislocated assemblies are then loaded into the each of the identified locations and the cycle is depleted as noted above.

The ΔCPR is then determined at each depletion step for each mislocated
assembly using the following equation:

$$\Delta CPR = SL \times \frac{MCPR_{Base} - MCPR_{ML}}{MCPR_{ML}} \qquad 5.3.5\text{-}1$$

where SL is the fuel cladding integrity limit MCPR, $MCPR_{Base}$ is the core wide MCPR for the final design depletion and $MCPR_{ML}$ is the core wide MCPR for the mislocated case. The largest ΔCPR from all the mislocated assemblies at all depletion steps is selected as the mislocated ΔCPR.

5.3.5.5 Sensitivity Studies/Justification of Procedure

The fuel loading error - mislocated fuel assembly analysis provides the maximum consequences for any loading error in the core. No further uncertainties are considered.

5.3.5.6 Typical Results

From the method described in Section 5.3.5.4, analysis was performed for a mislocated assembly in Cycle 4. In Table 5.3.5-1, the following cases are identified. Case 1 has a once-burned high reactivity assembly at a high CPR location. Case 2 has a high reactivity assembly face adjacent to a low CPR location. Case 3 has a fresh high reactivity assembly at a high CPR location. Case 4 has a fresh high reactivity assembly face adjacent to a low CPR position. Case 5 has a low reactivity assembly at a low CPR position next to a LPRM location. Case 6 has a once-burned high reactivity assembly next to a low CPR position with all control rods out for both the base case and the mislocated case. Case 7 has a fresh high reactivity assembly at a high CPR location with all control rods out for both the base case and the mislocated case.

As can be seen from Table 5.3.5-1, the worst mislocated assembly for Cycle 4 using the Supply System codes and the ANF, CPR correlation ANFB gives a value of ΔCPR of 0.10. Using ANF's results for Cycle 4, one obtains a ΔCPR of 0.14 for the mislocated assembly. The ANF results were obtained using the previously approved methodology for a mislocated assembly and the XN-3 CPR correlation. For Cycle 7, ANF using the currently approved methodology for the mislocated assembly and the ANFB correlation got a value of 0.12 for the worst ΔCPR. As can be seen from results of the other event analyses, this event is not limiting for the cycles evaluated. However, it will continue to be evaluated each cycle as necessary.

Table 5.3.5-1

ΔCPR for Mislocated Fuel Assembly

Exposure (GWD/MT)	ΔCPR Case 1	ΔCPR Case 2	ΔCPR Case 3	ΔCPR Case 4	ΔCPR Case 5	ΔCPR(ARO) Case 6	ΔCPR(ARO) Case 7
0.2	0.10	0.06	0.02	0.03	0.01	0.04	0.00
0.6	0.10	0.08	0.05	0.04	0.02	0.04	0.01
1.2	0.05	0.07	0.04	0.04	0.02	0.03	0.04
1.8	0.06	0.06	0.05	0.07	0.02	0.01	0.08
2.4	0.01	0.03	0.04	0.08	0.01	0.00	0.09
3.0	0.00	0.04	0.07	0.09	0.03	0.00	0.10
3.6	0.00	0.04	0.03	0.05	0.01	0.00	0.10
4.2	0.00	0.04	0.00	0.05	0.01	0.00	0.10
4.8	0.00	0.04	0.01	0.06	0.02	0.00	0.10
5.5	0.00	0.04	0.09	0.10	0.02	0.00	0.10

5.3.6 Fuel Loading Error--Rotated Fuel Assembly

5.3.6.1 Event Description

The rotated fuel assembly fuel loading error is the postulated occurrence of the loading of a fuel assembly in an improper orientation (rotated 90° or 180° from its intended orientation). This causes a discrepancy between the design core configuration and the actual core configuration. It is assumed that the loading error is not detected and that the plant operates throughout the cycle as constrained by the core operating limits for the design core configuration. Two errors must take place for this event to occur. A fuel assembly must be loaded in the wrong orientation and the loading error must be overlooked during the final core configuration verification.

The consequences of the rotated fuel assembly fuel loading error are dependent upon the lattice design. WNP-2 is a C lattice core design with uniform water gaps between fuel assemblies and essentially symmetrical fuel rod enrichments adjacent to all water gaps. Because of the spacer buttons at the top of the channel, a rotated fuel assembly will be slightly tilted. This causes the water gap thickness outside of the channel, and therefore, the local power distribution within the fuel assembly, to vary axially. The power distribution change will affect the actual CPR of the rotated fuel assembly. The core monitoring system assumes correct assembly orientation so the calculated thermal margin will be incorrect.

Because of the low probability of this event, it is considered an accident in the safety analysis process, and no other events or equipment failures are assumed to occur while the plant is operating with a fuel assembly loading error. However, the fuel cladding integrity limit has been adopted as the figure of merit for this event.

5.3.6.1.1 Initial Conditions and Operational Assumptions

The following plant operational conditions and assumptions form the principal bases for the analysis of the fuel loading error - rotated fuel assembly event:

(a) A rotated fuel assembly fuel loading error with the worst consequences has occurred and gone undetected.

(b) The plant is operating at rated conditions.

(c) The remaining NSSS operating parameters are consistent with normal plant operation.

(d) All of the plant control systems function normally.

5.3.6.1.2 Operator Actions

The operator is assumed to follow normal plant operating procedures because there is no information to indicate the existence of a fuel assembly loading error.

5.3.6.1.3 Event Acceptance Limits

The acceptance limit for this event is MCPR $\geq$ fuel cladding integrity safety limit. Compliance with this limit is demonstrated by assuring that the operating limit MCPR is greater than or equal to the MCPR which assures that greater than 99.9% of the fuel rods in the core are not expected to experience boiling transition plus the change in ΔCPR.

5.3.6.2 Analysis Considerations

This section describes the key analysis considerations applicable to the fuel loading error - rotated fuel assembly event. It includes: (1) a description of the phenomena occurring during the event that have a significant impact on the event consequences; (2) a discussion of the system performance characteristics that can significantly affect the course of the event; and (3) a discussion

of the performance characteristics of the important components as they relate to the event consequences.

5.3.6.2.1 Key Phenomena

This event is the insertion of a fuel assembly in its correct location but rotated 90° or 180° from its proper orientation. The fuel assembly rotation can result in a significant change in the local power distribution within the rotated assembly and a slight increase in the fuel assembly power. The water gap surrounding the rotated fuel assembly minimizes its impact on neighboring fuel assemblies. Therefore, only the phenomena associated with the rotated assembly and the feedback effects from neighboring assemblies are important for the analysis of the event.

5.3.6.2.2 Systems Considerations

All systems are assumed to be in normal power operational modes and continue to operate throughout the operating cycle.

5.3.6.2.3 Component Performance Characteristics

Because of the possibility of a rotated fuel assembly in an unmonitored location, it is assumed that it will not be detected by the core monitoring system. Therefore, control rod patterns may be selected that would place a correctly loaded fuel assembly on thermal limits which results in the rotated fuel assembly exceeding normal operating thermal limits.

5.3.6.3 Methodology/Integration of Codes and Analysis

The fuel loading error - rotated fuel assembly utilizes the CASMO-2E lattice physics code as the primary analysis codes. CASMO-2E is used to calculate the local peaking factors for both the correctly loaded fuel assembly and the rotated fuel assembly. The MICBURN-E code is used to determine the gadolinia cross

sections used in CASMO-2E and ESCORE provides the fuel temperature distribution. FICE is used to determine the local peaking functions for both the correctly load assembly and the rotated assembly for the thermal limits evaluation. TLIM is used for the thermal limits evaluation. (See Section 2 and Figure 2-1 for an overview of the overall WNP-2 reactor analysis methodology computer code sequence.)

5.3.6.4 Licensing Analysis Procedure

The rotated assembly analysis is performed on the basis of CASMO-2E lattice physics calculations of the local peaking factors for the correctly oriented and rotated assembly using the final design depletion through the cycle with control rods present.

The rotated assembly CASMO-2E calculation is based the displacement of the assembly due to the spacer buttons at the top of the channel. For conservatism, it is assumed that the whole assembly is displaced by the same amount when in actuality the assembly will be tilted with only the top of the assembly being displaced by the full amount.

The CPR calculation for the rotated assembly is then performed for the limiting assembly types in terms of local peaking and reactivity through the cycle. The ΔCPR is determined using the following equation:

$$\Delta CPR = SL \times \frac{MCPR_{Base} - MCPR_{Rot}}{MCPR_{Rot}} \qquad 5.3.6\text{-}1$$

where fuel cladding integrity limit MCPR, $MCPR_{Base}$ is the core wide MCPR for the final design depletion and $MCPR_{Rot}$ is the core wide MCPR for the rotated assembly case.

The largest ΔCPR from the worst possible rotations at all exposures in the cycle is selected as the rotated assembly ΔCPR.

5.3.6.5 Sensitivity Studies/Justification of Procedure

In the licensing analysis procedure, it is assumed that the maximum water gap exists over the full length of the rotated fuel assembly. This is a very conservative assumption that dominates any uncertainties in the analysis process. If it is determined that the assumption of the maximum water gap leads to core operating limits that are unnecessarily restrictive, analyses will be performed using a variable water gap to reduce the conservatism in the analysis process. If the variable water gap approach is taken, appropriate uncertainty analyses will be performed to establish their impact on the analysis results.

5.3.6.6 Typical Results

From the method described in Section 5.3.6.4, analysis was performed for a rotated assembly in Cycle 4. Table 5.3.6-1 shows the results for the highest ΔCPR at each exposure point in the cycle.

As can be seen from Table 5.3.6-1, the highest ΔCPR for a rotated assembly in Cycle 4 is 0.15. The ANF Corporation does not perform a rotated assembly analysis for WNP-2. As can be seen from results of the other event analyses, this event is not limiting for the cycles evaluated. However, it will continue to be evaluated each cycle as necessary.

Table 5.3.6-1

Rotated Assembly ΔCPR

Exposure (GWD/MT)	ΔCPR
0.2	0.13
0.6	0.15
1.2	0.11
1.8	0.11
2.4	0.12
3.0	0.11
3.6	0.09
4.2	0.10
4.8	0.11
5.5	0.12

5.3.7 Feedwater Controller Failure--Maximum Demand

5.3.7.1 Event Description

The feedwater controller failure (FWCF) transient event is initiated by the failure of a control device which results in the feedwater controller being forced to its upper limit which creates the maximum feedwater system flow demand. The increased feedwater flow mixes with the recirculation flow and results in a gradual increase in core inlet subcooling. The increased feedwater flow also results in an increase in reactor vessel water level. The gradual increase in core inlet subcooling causes a relatively slow power increase and a shift in power distribution towards the bottom of the core. As a result of the power increase, the vessel steam flow increases which results in a slight increase in system pressure due to the larger steam line pressure drops as the pressure regulator system controls the turbine inlet pressure. The power increase continues until the reactor vessel high water level trip setpoint (L8) is reached.

High reactor vessel water level initiates closure of the main turbine stop valves (turbine trip) and a trip of the feedwater system. Closure of the turbine stop valves initiates a reactor scram, a bypass valve opening signal, and prompt RPT. Following the turbine trip, the neutron flux increase is limited by the reactor scram and the prompt RPT. The peak neutron flux and surface heat flux are reached following the turbine trip. The relief valves are opened in the pressure relief mode and close sequentially as the pressure is reduced by the action of the relief valves and the turbine bypass valves.

Table 5.3.7-1 shows the expected sequence of events for the feedwater controller failure - maximum demand transient.

5.3.7.1.1 Initial Conditions and Operational Assumptions

The following plant operational conditions and assumptions form the principal bases for the analysis of the feedwater controller failure - maximum demand event:

(a) The plant is operating at the safety analysis power level and rated core flow.

(b) The remaining NSSS operating parameters are consistent with normal plant operation.

(c) The feedwater controller fails during maximum flow demand resulting in maximum feedwater pump runout.

(d) The plant is operating in the manual flow control mode.

(e) The system trips and initiation signals are consistent with technical specifications.

(f) High reactor pressure vessel water level trips the main turbine stop valves and the feedwater pumps.

(g) Position switches on the turbine stop valve initiate reactor scram, and prompt RPT. The turbine bypass valve receive their opening signal from the pressure regulator system upon receipt of the turbine trip signal.

(h) All of the remaining plant control systems function normally.

5.3.7.1.2 Operator Actions

No restart is assumed and the reactor is to be cooled down. The operator is expected to take the following actions as appropriate:

(a) Observe that the high water level feedwater pump trip and scram has terminated the event.

(b) Switch the feedwater controller from automatic to manual control in order to try to regain a correct output signal.

(c) Identify causes of the failure and report all key plant parameters during the event.

(d) Cool down the reactor consistent with plant procedures.

5.3.7.1.3 Event Acceptance Limits

The acceptance limits for this event are MCPR $\geq$ fuel cladding integrity limit; LHGR $\leq$ PAFF limits; and reactor pressure $\leq$ the ASME Code limit for the reactor coolant pressure boundary. Compliance to the fuel cladding integrity limit is demonstrated by assuring that the operating limit MCPR is greater than or equal to the fuel cladding integrity MCPR limit (which assures that greater than 99.9% of the fuel rods in the core are not expected to experience boiling transition) plus the change in ΔCPR during the event. Compliance to the PAFF limit is assured by meeting the LHGR limit requirements for transient occurrences in the fuel vendor mechanical design topical reports (ex. Reference 16).

5.3.7.2 Analysis Considerations

This section describes the key analysis considerations applicable to the feedwater controller failure - maximum demand event. It includes: (1) a description of the phenomena occurring during the event that have a significant impact on the event consequences; (2) a discussion of the system performance characteristics that can significantly affect the course of the event; and (3) a discussion of the performance characteristics of the important components as they relate to the event consequences.

5.3.7.2.1 Key Phenomena

Described below are the key phenomena related to the feedwater controller failure - maximum demand event. Consideration of these phenomena is necessary in the simulation of this event in order to accurately model the plant response.

The feedwater controller failure is a relatively complex event because of its two distinct phases. The first phase is similar to a core coolant temperature decrease transient where the change in inlet subcooling due to the increased feedwater flow results in a relatively slow change in the key operating parameters. The major

complexity during this phase is modeling of the core power and the power distribution shifts. The second phase is very dynamic and is characterized by rapidly changing conditions with complex interactions similar to a turbine trip with bypass from a higher power level.

<u>Feedwater Phenomena</u> The event begins with a rapid increase in feedwater flow. As the additional feedwater is mixed with the recirculation flow, the core inlet subcooling is increased which increases core power. Because the duration of this phase is short compared to the time constant of the feedwater system, changes in feedwater heating are neglected. Therefore, the feedwater phenomena are modeled as a step increase in feedwater flow to the vessel until the high reactor vessel water level trip point is reached. Following the feedwater system trip, the feedwater flow coasts down to zero for the remainder of the event.

<u>Steam Line Phenomena</u> During the initial phase of the event, the steam flow increases as the core power increases. The turbine inlet and bypass valves open to maintain turbine inlet pressure. The steam line pressure slowly increases.

The final phase of the event is initiated by the reactor vessel high water level trip of the turbine stop valves and feedwater flow. The turbine stop valve closure causes a pressure increase at the turbine inlet that is rapidly transmitted to the reactor pressure vessel by pressure wave phenomena in the steam lines. The relief valves open at pre-established setpoints allowing a steam release path for pressure relief. Nodalization of the steam lines is necessary to assure accurate simulation of the system pressure response.

<u>Pressure Vessel Phenomena</u> During the initial phase of the event, the additional feedwater flow enters the reactor pressure vessel and is mixed with the recirculation flow in the downcomer annulus. In the final phase of the event, the propagation of the pressure wave from the steam lines to the core is an important phenomenon.

The attenuation of the pressure wave by the reactor vessel internal components is a particularly important phenomenon in modeling the timing of the core moderator void changes.

Recirculation Phenomena The flow control valve remains in a fixed position. Following the turbine trip, the reduced core flow, due to prompt RPT, will help increase core steam voids which helps to mitigate the reactivity insertion due to the pressure increase. The recirculation system modeling includes consideration of the downcomer phenomena, including the annular flow region above and through the jet pumps. The changing performance of the jet pumps at varying pressure and drive flow is included. The external recirculation loop flow is represented so that flow into the vessel as a function of time is accurately simulated.

Core Phenomena The important phenomena in the core region are the reactivity effects that contribute to changes in the neutron flux and fuel surface heat flux.

During the initial phase of the transient, the primary phenomena are the effects of the increase in inlet subcooling on the moderator temperature and void reactivity.. The change in moderator temperature and the reduction in steam voids due to the colder water entering the core is responsible for the initial positive reactivity insertion. This causes an increase in power level and fuel temperature which increases the Doppler reactivity and generates additional steam voids to mitigate the effects of the decrease in core inlet temperature. The net effect is a slow rise in core power and a shift in the power distribution toward the bottom of the core.

During the second phase of the transient, following the turbine trip, the steam void reactivity, due to steam void collapse as the pressure wave from the turbine inlet reaches the core, provides the additional positive reactivity insertion. The reactivity increase causes a rapid increase in core power, which then increases fuel temperature, core voids, and system pressure. However, with steam flow restricted, system pressure and core power continue to increase until a rapid scram of the control rods and trip of the recirculation pump is initiated by the turbine stop valve closure (valve position switch). During this portion of the event, the time and spatial dependent scram reactivity effects are simulated.

5.3.7.2.2 Systems Considerations

For the feedwater controller failure transient, the initiating event is a failure which results in the maximum demand signal to the feedwater system. All other systems normally operating are assumed to function as designed. Safety systems that are designed to actuate are assumed to actuate at their pre-established setpoints. The steam bypass is assumed to operate as designed.

The important systems to be considered are: (1) the reactor protection system including the turbine stop valve position scram; (2) the control rod drive (scram) system; (3) the steam system including stop valves, bypass valves, and relief valves; (4) the recirculation system, including the prompt RPT; (5) the steam separation system inside the vessel; (6) the fuel and core system, including the nuclear/thermal-hydraulic coupling; and (7) feedwater control system. Other systems called upon for long-term operation are not required to be part of this analysis because their action occurs much later in the transient following the time of challenge or nearest approach to the event acceptance limits.

5.3.7.2.3 Component Performance Characteristics

The feedwater controller failure transient analysis requires detailed modeling of the NSSS in order to assure that all systems

that influence reactor system pressure, steam flow, core flow, and core inlet enthalpy are properly considered. The selection of licensing basis component performance characteristics is based on a buildup of conservative assumptions established by past practices and licensing requirements.

Feedwater System The characteristics of the feedwater system are important for the feedwater controller failure transient. The largest increase (pump runout) in feedwater flow from the operating state produces the most severe event consequences. The maximum feedwater flow rate has been compared to plant startup test results and current system performance.

Turbine Stop Valve Closure Characteristics The turbine stop valve closure characteristics are important in the period following the high reactor vessel level trip of the main turbine. The turbine stop valve characteristics are specified by the turbine manufacturer and confirmed during plant startup testing. The technical specifications establish the value for closure time to the trip setpoints for the reactor protection system. From that position to full closure, a linear ramp change in valve position is assumed.

Steam Bypass Valve Opening Characteristics The steam bypass valve opening characteristics are important in determining the steam system pressure and flow. The steam bypass valve opens on signals from the pressure regulator system upon receipt of the turbine trip signal. The opening time is specified in the design process and confirmed during plant startup testing. A conservative time delay for bypass valve opening is applied to the licensing basis analysis.

Relief Valve Characteristics The relief valves are used to protect the reactor coolant pressure boundary against overpressure events. The technical or design specifications establish limiting conditions for the relief valve setpoints. The maximum values are

used in the licensing basis analysis in order to assure conservative evaluation of the system pressure response.

Recirculation Pump Coastdown Characteristics The recirculation pumps receive a trip signal from the turbine stop valve closure which trips a breaker to the recirculation pump power supply. The slowest pump coastdown consistent with design specifications is used in the licensing basis analysis.

Reactor Protection System Signal Delays The reactor protection system includes the collection of a number of analog and digital signals, conditioning of these signals, comparison to pre-established setpoint limits and activation of nuclear system trips. The signal processing and trip initiation involves delay times which impact transient response. The plant technical specifications identify the allowable reactor protection system response times.

Control Rod Drive Insertion Time The control rod drive system provides the primary mechanism for negative reactivity insertion for terminating the transient. The control rod drives are inserted in the scram mode by the scram hydraulic control system. The scram time for the control rods is consistent with the technical specification surveillances.

High Reactor Water Level (L8) Trip The high reactor water level trip initiates the closure of the turbine stop valves and the feedwater system trip. The maximum setpoint identified in the technical specification is used to maximize the event dynamics.

5.3.7.3 Methodology/Integration of Codes and Analysis

The primary analysis model in the simulation of the feedwater controller failure - maximum demand is the system thermal hydraulic model, RETRAN-02. RETRAN-02 is used to calculate the changes in system and core average nuclear and thermal hydraulic parameters throughout the course of the event. The RETRAN-02 analysis results

are used in the assessment of fuel thermal margin, the increase in nodal power, and the peak reactor vessel pressure.

The analysis of the feedwater controller failure maximum demand is performed using the following codes in the sequence shown on Figure 2-1: (1) ESCORE; (2) MICBURN-E; (3) CASMO-2E; (4) NORGE-B; (5) SIMULATE-E; (6) SIMTRAN-E; (7) STRODE; (8) RETRAN-02; (9) FICE; (10) VIPRE-01; and (11) STARS (when limiting). ESCORE is used to provide the fuel rod temperature distribution used in CASMO-2E and the gap conductance used in RETRAN-02 and VIPRE-01. MICBURN-E provides the gadolinia cross sections used in CASMO-2E. CASMO-2E is used to perform the lattice physics analysis to generate the cross sections for SIMULATE-E and the inverse neutron velocity and total effective delayed neutron yield for SIMTRAN-E. NORGE-B is used to transfer the CASMO-2E data to SIMULATE-E and SIMTRAN-E. SIMULATE-E develops the three-dimensional macroscopic cross section data to be processed by SIMTRAN-E. SIMTRAN-E collapses the three-dimensional cross section data to one dimension and transfers the other nuclear parameters to RETRAN-02. STRODE is used adjust the moderator density feedback behavior and delayed neutron fraction data for input to RETRAN-02. RETRAN-02 is used to perform the transient analysis. VIPRE-01 is used to determine the ΔCPR during the transient based on the local peaking function provided by FICE. STARS, if required, is used to perform the statistical assessment to demonstrate compliance with the fuel cladding integrity or PAFF limits.

5.3.7.4 Licensing Analysis Procedure

In the analysis of the feedwater controller failure - maximum demand, the following conditions are applied:

(a) The rod insertion times are based on plant surveillance data and technical specification requirements.

(b) Scram is initiated by the position switches on the turbine stop valves. Should a high neutron flux or high thermal power trip due to the power increase during the initial phase of the event be predicted to occur, it is conservatively ignored. Scram time delay is the maximum technical specification value.

(c) RPT time delay is the maximum technical specification value. An analysis is also performed without RPT to establish the appropriate operating limit for plant operation with RPT out of service.

(d) Relief valve opening setpoints are maximum valves consistent with technical specifications.

(e) The analysis is performed at end of cycle conditions, with all control rods fully withdrawn.

(f) The analysis is performed at the most limiting point on the power/flow operating map, consistent with the license basis assumption on maximum power level. An analysis is also performed at 47% power and 106% flow.

(g) Feedwater temperature is determined by the RETRAN code consistent with the system heat balance at the licensing power level. To support plant operation with final feedwater temperature reduction to extend the operating cycle, the limiting feedwater temperature is calculated using a consistent set of nuclear parameters.

(h) The turbine stop valves have a full stroke closure time, from fully open to fully closed, of 0.07 seconds.

(i) A bypass valve opening time delay is applied to provide conservatism in the licensing basis analysis.

(j) The plant is operating in the manual flow control mode.

(k) The feedwater flow is increased to 146% maximum flow at initiation of the transient.

Events caused by low reactor water level trip setpoints, including main steam line isolation valve closure and ECCS initiation, are not included in the simulation. These events, should they occur, will be later than the time of challenge or nearest approach to the event acceptance limits.

5.3.7.5 Sensitivity Studies/Justification of Procedure

RETRAN-02 and VIPRE-01 analyses were performed to determine the sensitivity of the calculated results to changes in input assumptions. The parameters considered in the sensitivity analysis were:

(a) Initial core power

(b) High level (L8) setpoint for turbine and feedwater trip

(c) Scram speed

(d) Prompt RPT

The parameter values used in the sensitivity studies and the calculated RCPR are shown in Table 5.3.7-2. The results of these studies are consistent. RPT mitigates the event consequences because of the reduced core flow and higher core average void fraction. Use of normal scram time also mitigates the event consequences because of earlier control blade negative reactivity insertion. Earlier turbine trip time mitigates the event consequences because the core power at the time of turbine trip is lower than in the base case.

5.3.7.6 Typical Results

This section shows the results of a WNP-2 Cycle 4 analysis of the feedwater controller failure - maximum demand event using the Supply System licensing analysis procedure.

The base case analyzed is 47% power/106% core flow with technical specification scram time and with RPT operable. Plots of core power, core average heat flux, core inlet flow, reactor vessel steam dome pressure, vessel steam flow, reactor vessel water level, and feedwater flow are provided on Figures 5.3.7-1 through 5.3.7-7, respectively.

Results of a high power case (104% power/100% core flow with technical specification scram time and with RPT operable) are also provided. Plots of core power, core average heat flux, core inlet flow, reactor vessel steam dome pressure, vessel steam flow, reactor vessel water level, and feedwater flow are shown on Figures 5.3.7-8 through 5.3.7-14, respectively.

A comparison of the Supply System and fuel supplier (ANF) calculated results for WNP-2 Cycle 4 is provided in Tables 5.3.7-3

and 5.3.7-4. For the 47% initial core power base case with technical specification scram time and RPT, the Supply System calculates RCPR of 0.280 compared to 0.107 calculated by ANF. Results are shown on Table 5.3.7-3. The Supply System calculates a significantly higher RCPR than the fuel vendor for this event. However; when the fuel vendor calculated turbine trip time is input to the Supply System calculation (rather than calculating the trip time from downcomer water level rise), the results are in good agreement. For the 104% initial core power case with RPT, the Supply System calculates RCPR of 0.179. Supply System results are shown on Table 5.3.7-4. ANF results for Cycle 4 are not available for this event.

Table 5.3.7-1

Feedwater Controller Failure - Maximum Demand Sequence of Events

Time (sec)	Events
0.0	Feedwater pumps runout to maximum flow.
17.59	High vessel water level (L8) setpoint trips feedwater pumps.
17.65	High vessel water level (L8) setpoint trips main turbine.
17.65	Turbine bypass operation is initiated.
17.66	Reactor scram trip is actuated by main turbine stop valve position switches.
17.82	Main turbine stop valves are fully closed.
17.85	The RPT is activated by main turbine stop valve position switches.
18.04	Recirculation pump motor circuit breakers open causing a decrease in core coolant flow.
19.19	Group 1 relief valves open to mitigate system pressure increase.
19.54	Group 4 relief valves open to mitigate system pressure increase.
*	Relief valves cycle closed/open to maintain system pressure.
**	Low level (L2) reactor vessel water level is reached.
**	Main steam line isolation valves close and high pressure coolant inventory makeup systems are initiated.

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* Group 4 closed at 22.12 seconds and Group 3 closed at 22.67 seconds.

** These events are beyond the RETRAN simulation times.

Table 5.3.7-2
Sensitivity Studies for
Feedwater Controller Failure - Maximum Demand

	RCPR	ΔCPR
Full Power Cases		
With RPT	0.179	0.22
No RPT	0.236	0.31
With RPT, normal scram time**	0.134	0.15
47% Power Cases		
With RPT	0.280	0.39
No RPT	0.319	0.47
With RPT, early turbine trip*	0.147	0.17
With RPT, normal scram time**	0.253	0.34

* Turbine trip time set equal to fuel vendor's calculation.

** Normal scram time is the rod insertion time based on WNP-2 measured values. The normal scram times are cited in the plant technical specifications and are approximately 2 standard deviations above the mean measured scram times. All other cases were performed with the technical specification scram speed which is a very conservative upper bound. A discussion of WNP-2 scram time is provided in Appendix A.

Table 5.3.7-3
Comparison of Supply System Results to ANF Results
for Feedwater Controller Failure - Maximum Demand
-- Low Power Case --

<u>Technical Specification Scram Time with RPT (47% power/106% flow)</u>

	Supply System Result	Fuel Vendor Result
Maximum Neutron Flux (% Rated)	88 (56*)	110
Maximum Core Average Heat Flux (% Rated)	63 (51*)	51
RCPR	0.280 (.147*)	0.107
ΔCPR	0.39 (0.17*)	0.12

* Turbine trip time set equal to fuel vendor's calculation

<u>Normal Scram Time with RPT (47% power/106% flow)</u>

	Supply System Result	Fuel Vendor Result
Maximum Neutron Flux (% Rated)	72	103
Maximum Core Average Heat Flux (% Rated)	61	50
RCPR	0.253	0.099
ΔCPR	0.34	0.11

Table 5.3.7-4
Comparison of Supply System Results to ANF Results for Feedwater Controller Failure - Maximum Demand
-- High Power Case --

Technical Specification Scram Time; RPT (104% power/100% flow)

	Supply System Result	Fuel Vendor Result
Maximum Neutron Flux (% Rated)	210	Not Available
Maximum Core Average Heat Flux (% Rated)	118	Not Available
RCPR	0.179	Not Available
ΔCPR	0.22	Not Available

Technical Specification Scram Time; no RPT (104% power/100% flow)

	Supply System Result	Fuel Vendor Result
Maximum Neutron Flux (% Rated)	328	Not Available
Maximum Core Average Heat Flux (% Rated)	129	Not Available
RCPR	0.236	Not Available
ΔCPR	0.31	Not Available

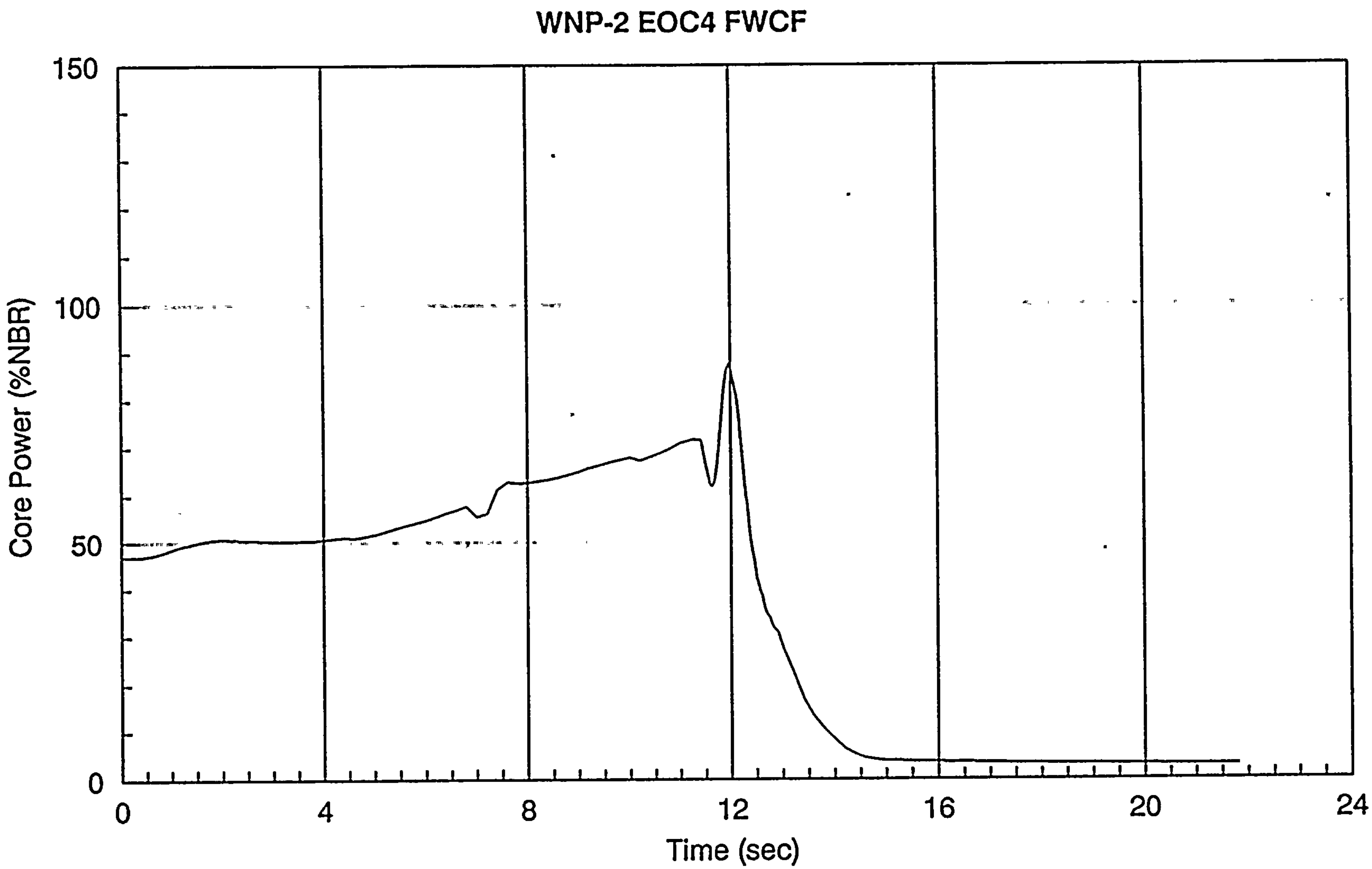

Figure 5.3.7-1 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

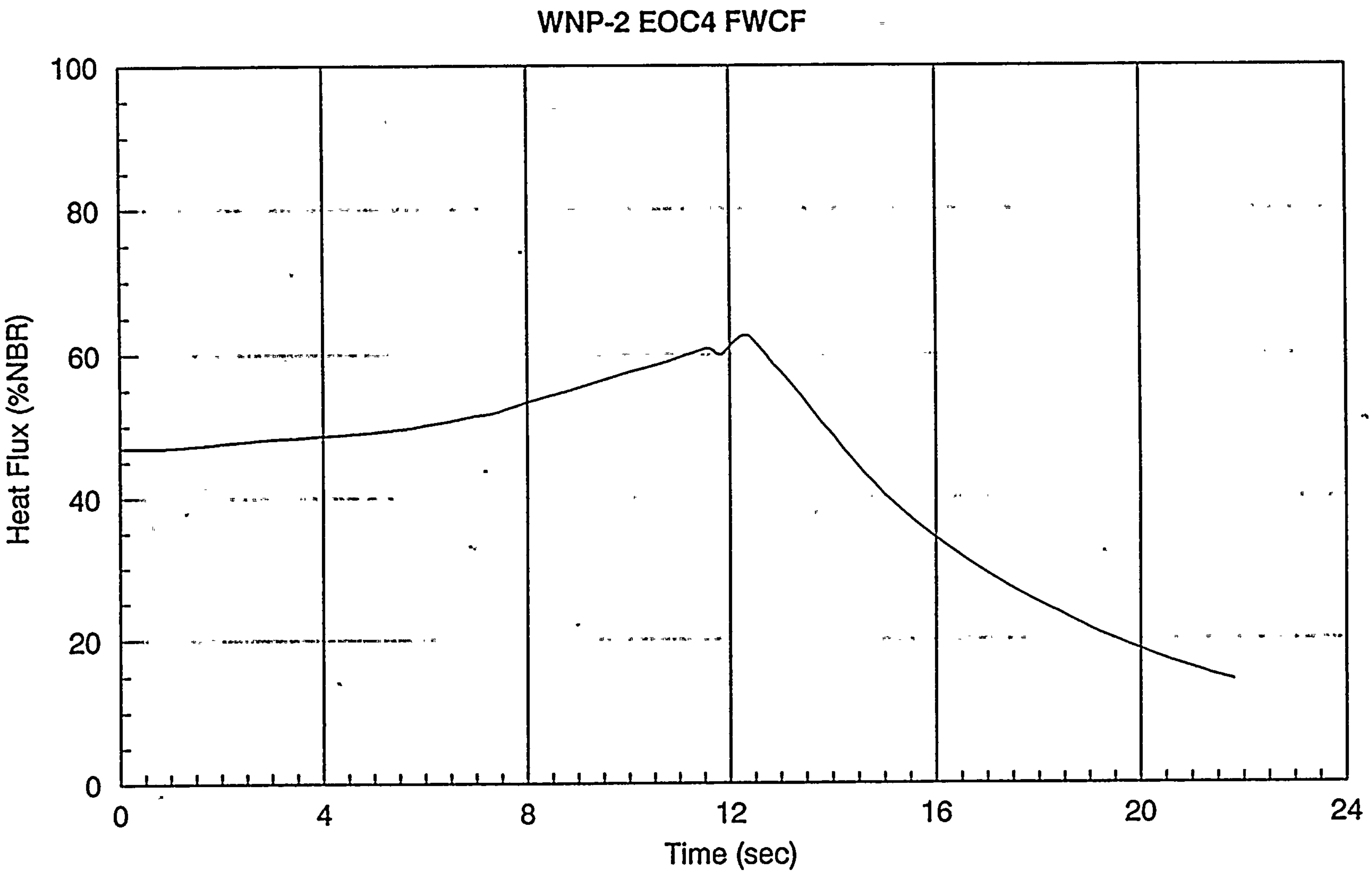

Figure 5.3.7-2 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

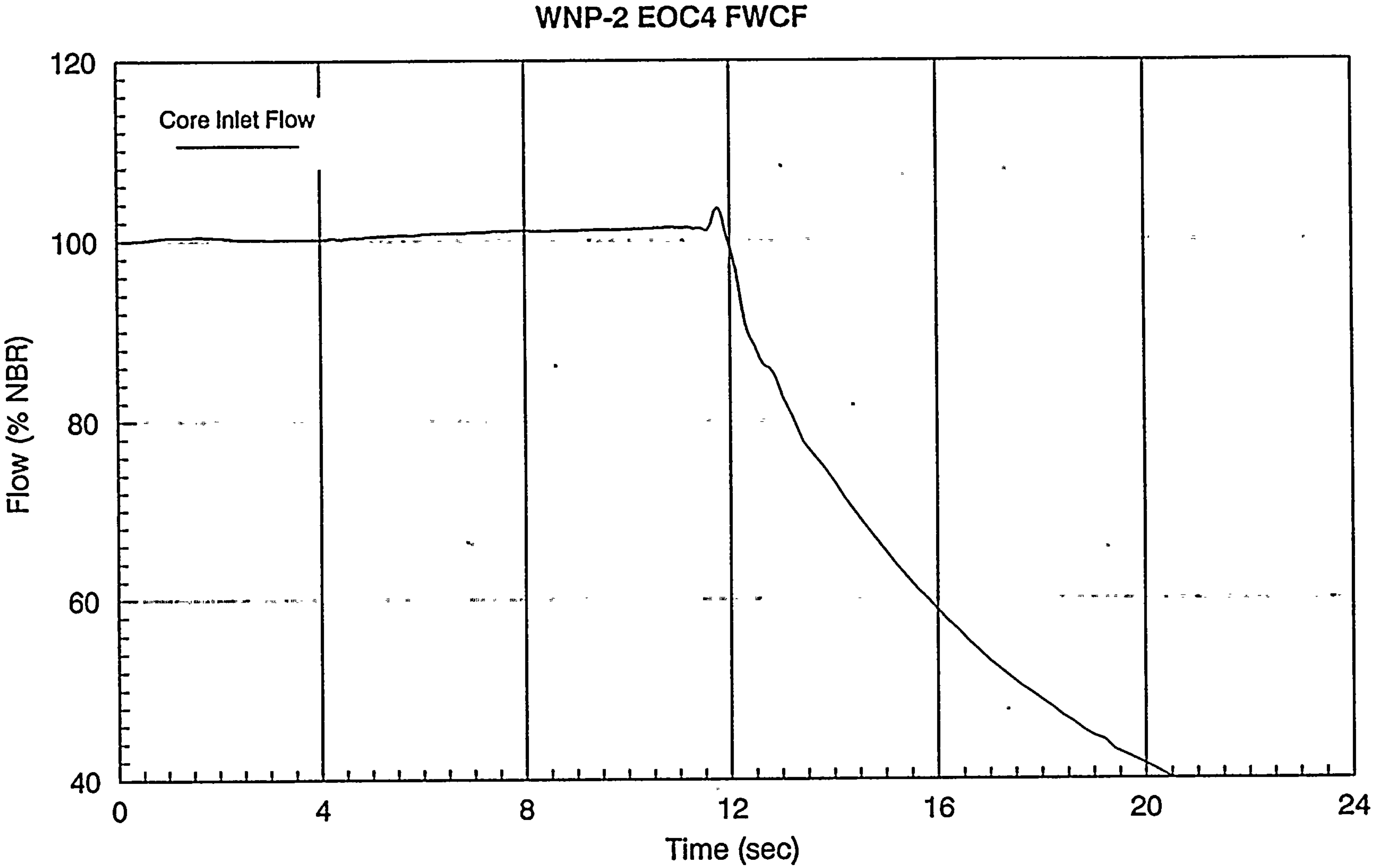

Figure 5.3.7-3 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

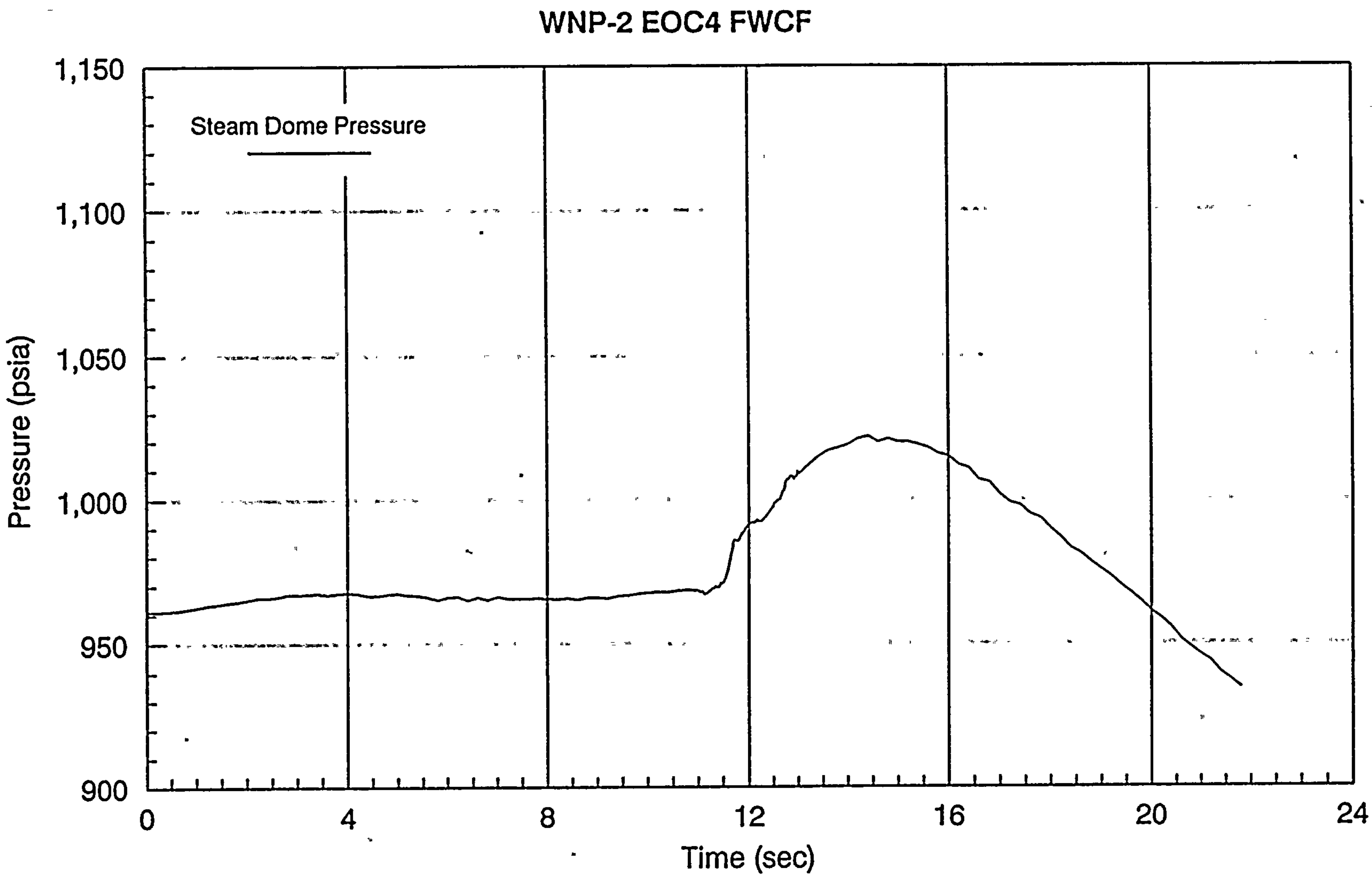

Figure 5.3.7-4 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

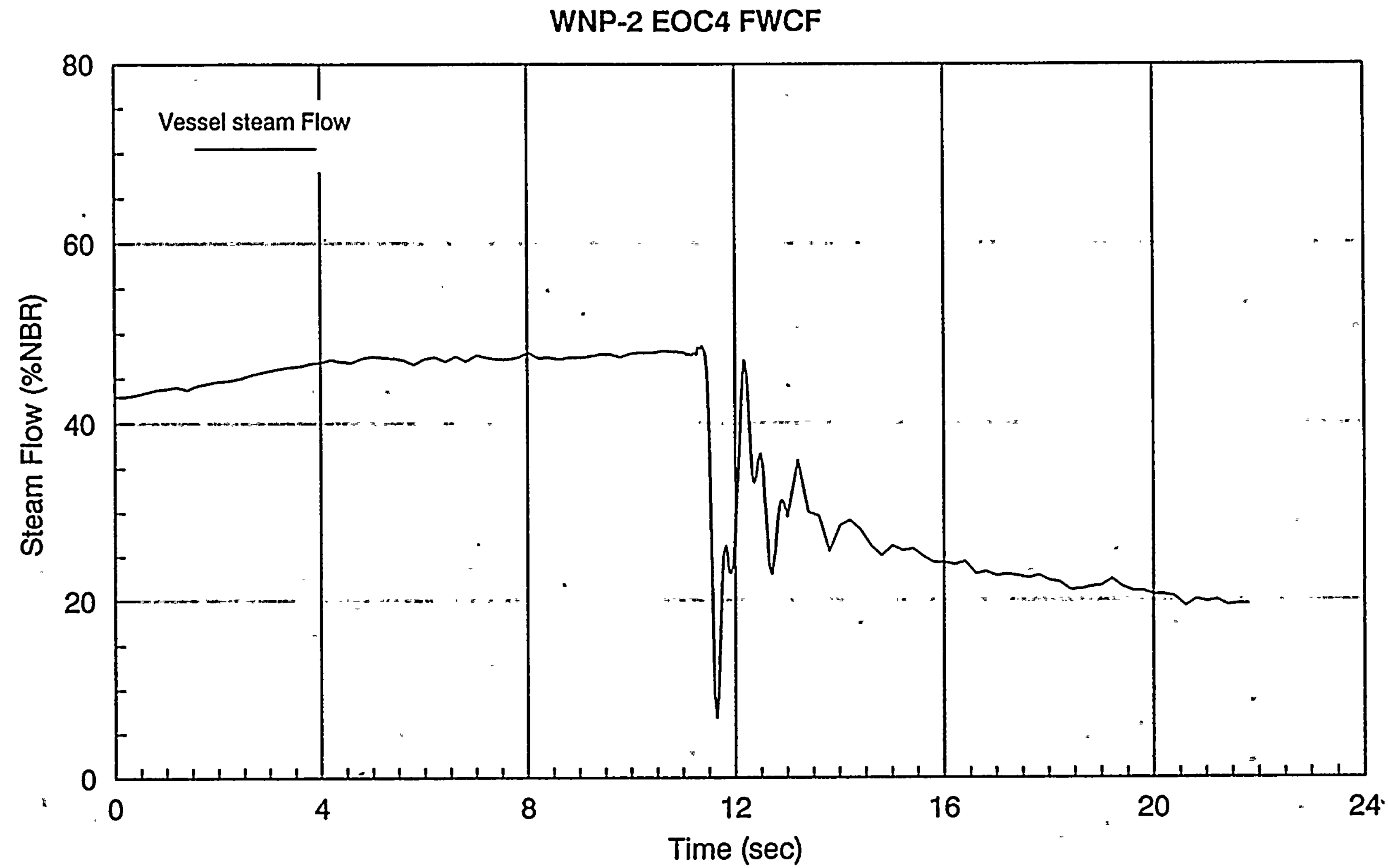

Figure 5.3.7-5 FWCF Resuts for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

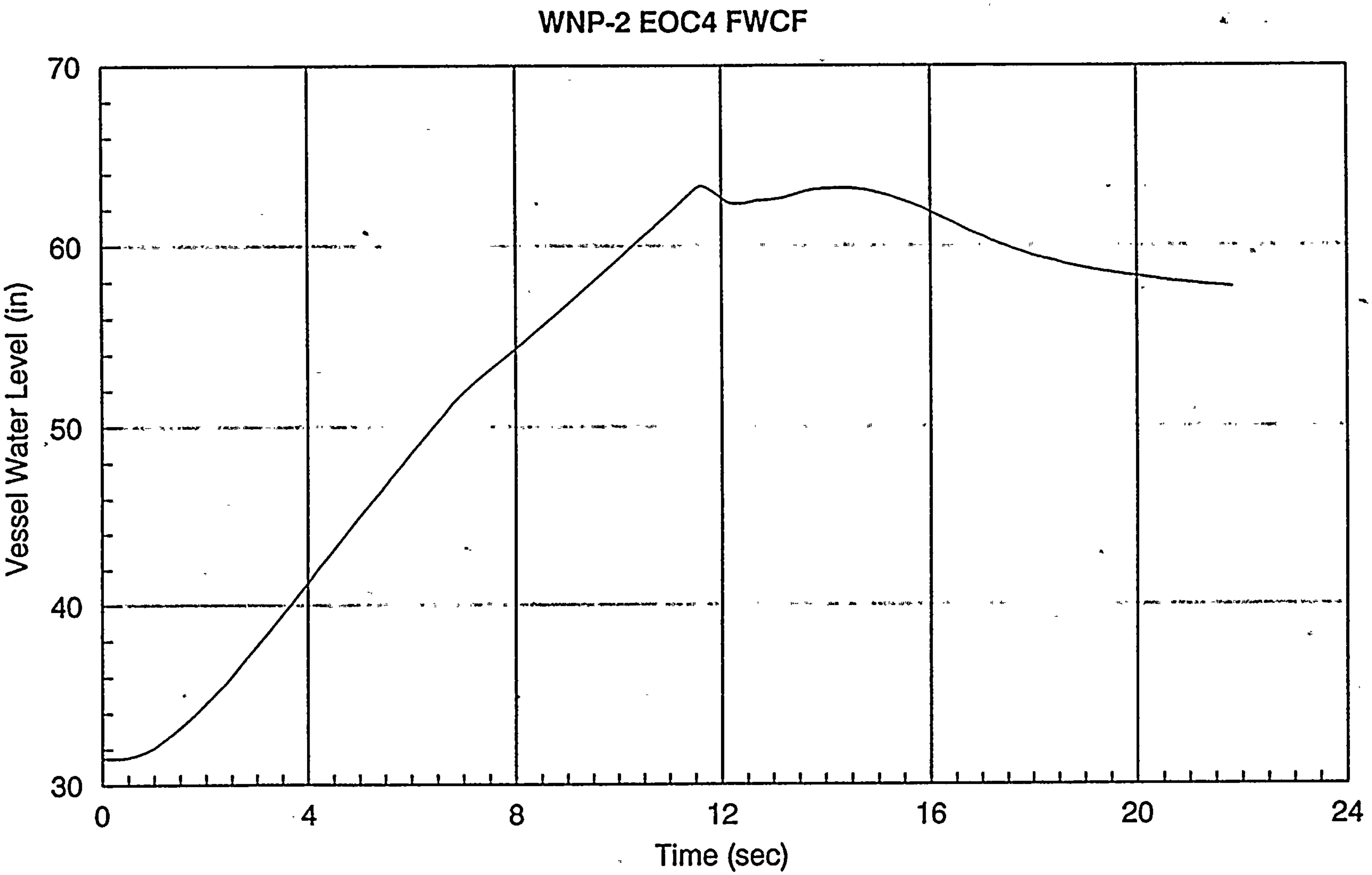

Figure 5.3.7-6 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

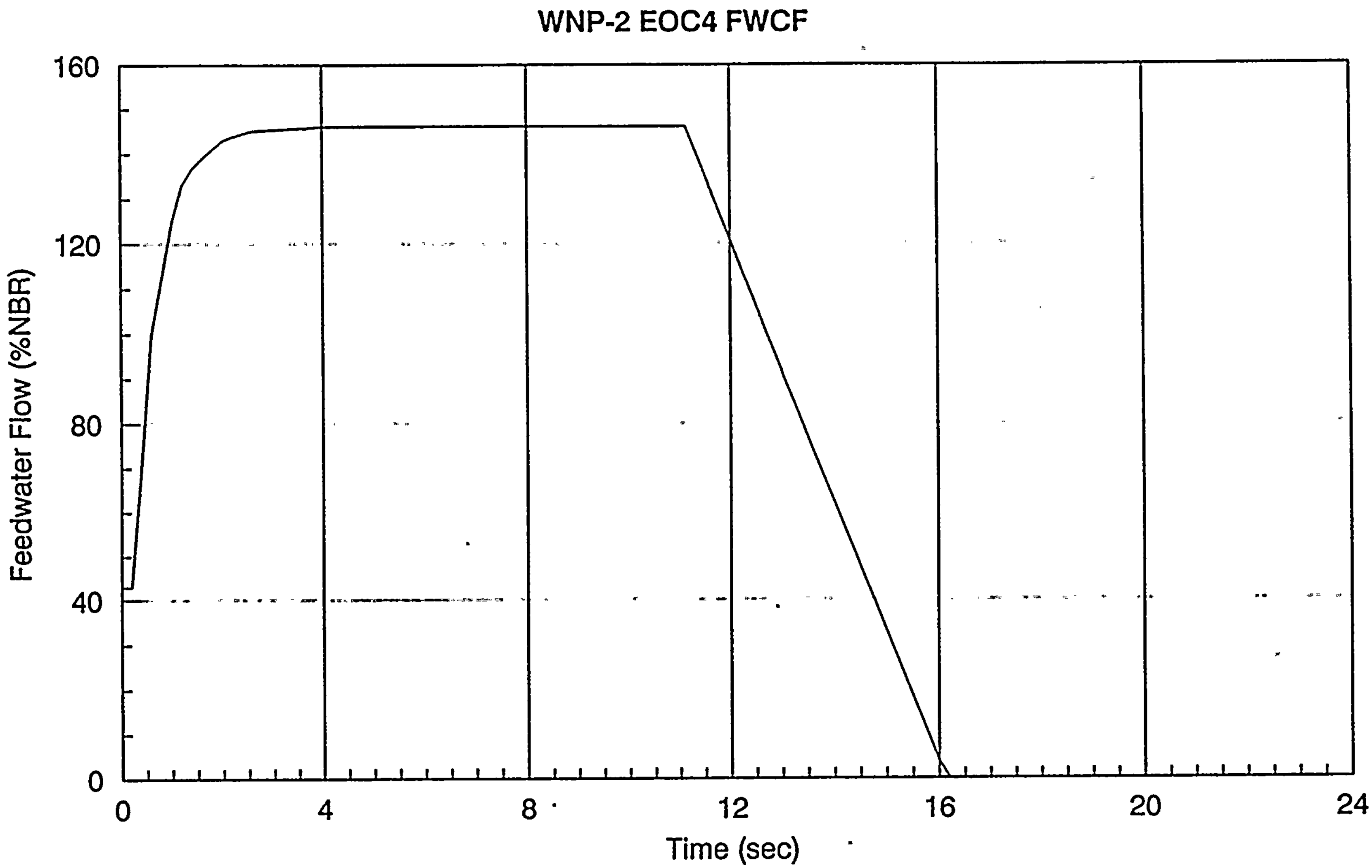

Figure 5.3.7-7 FWCF Results for 47% Power and 106% Flow, RPT Operable, Tech. Spec. Scram Time

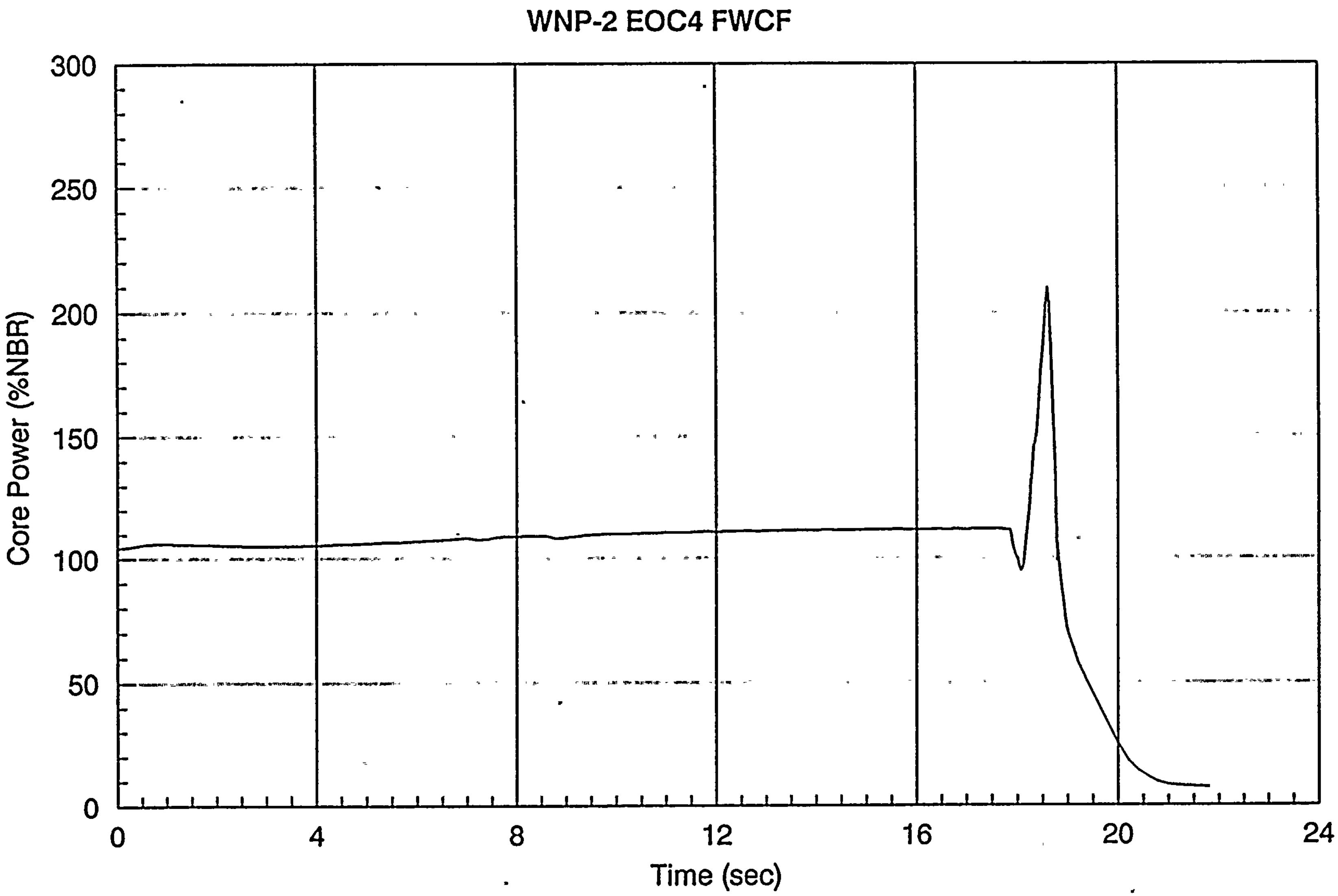

Figure 5.3.7-8 FWCF Results, RPT Operable, Tech. Spec. Scram Time

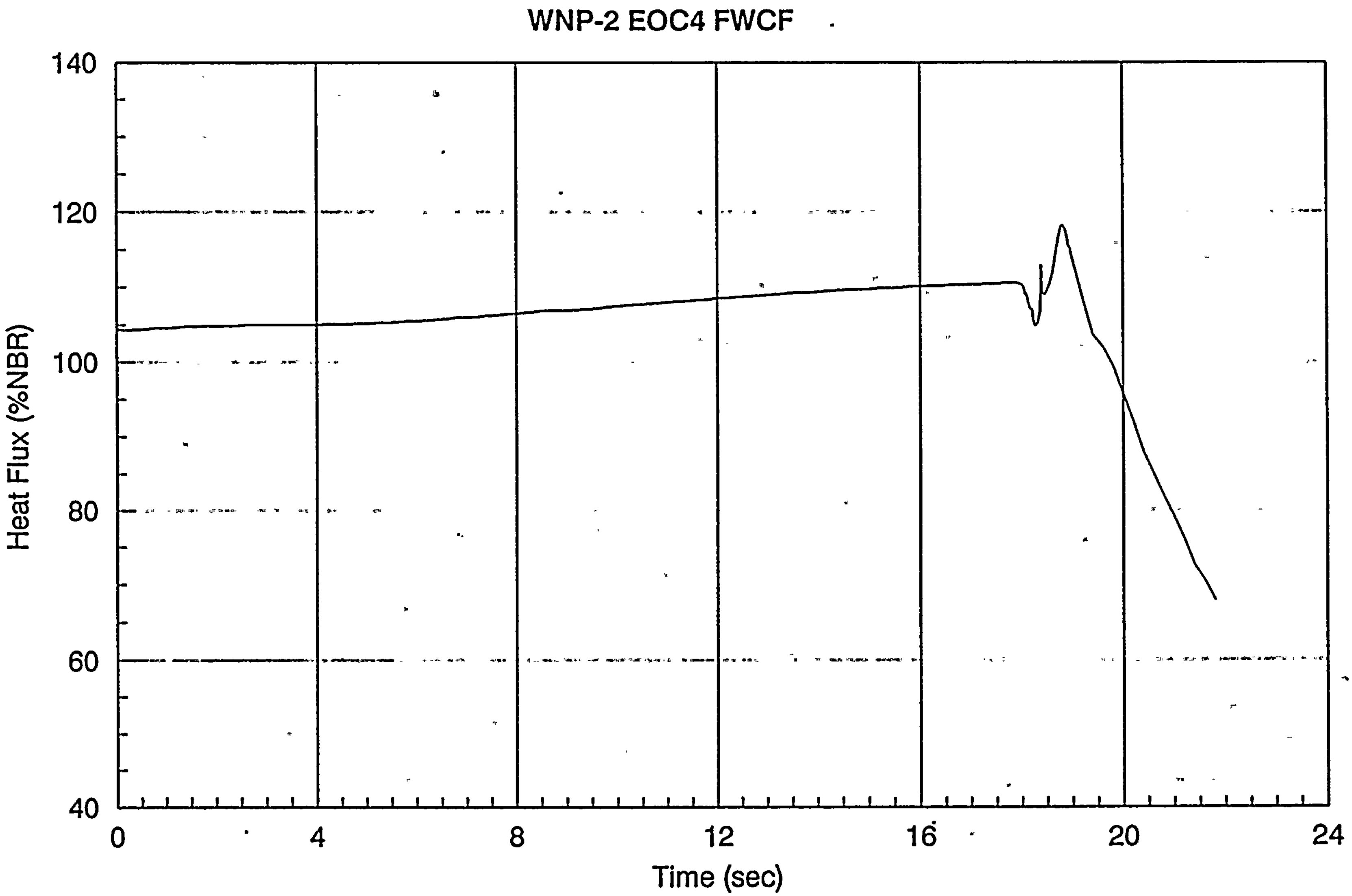

Figure 5.3.7-9 FWCF Results, RPT Operable, Tech. Spec. Scram Time

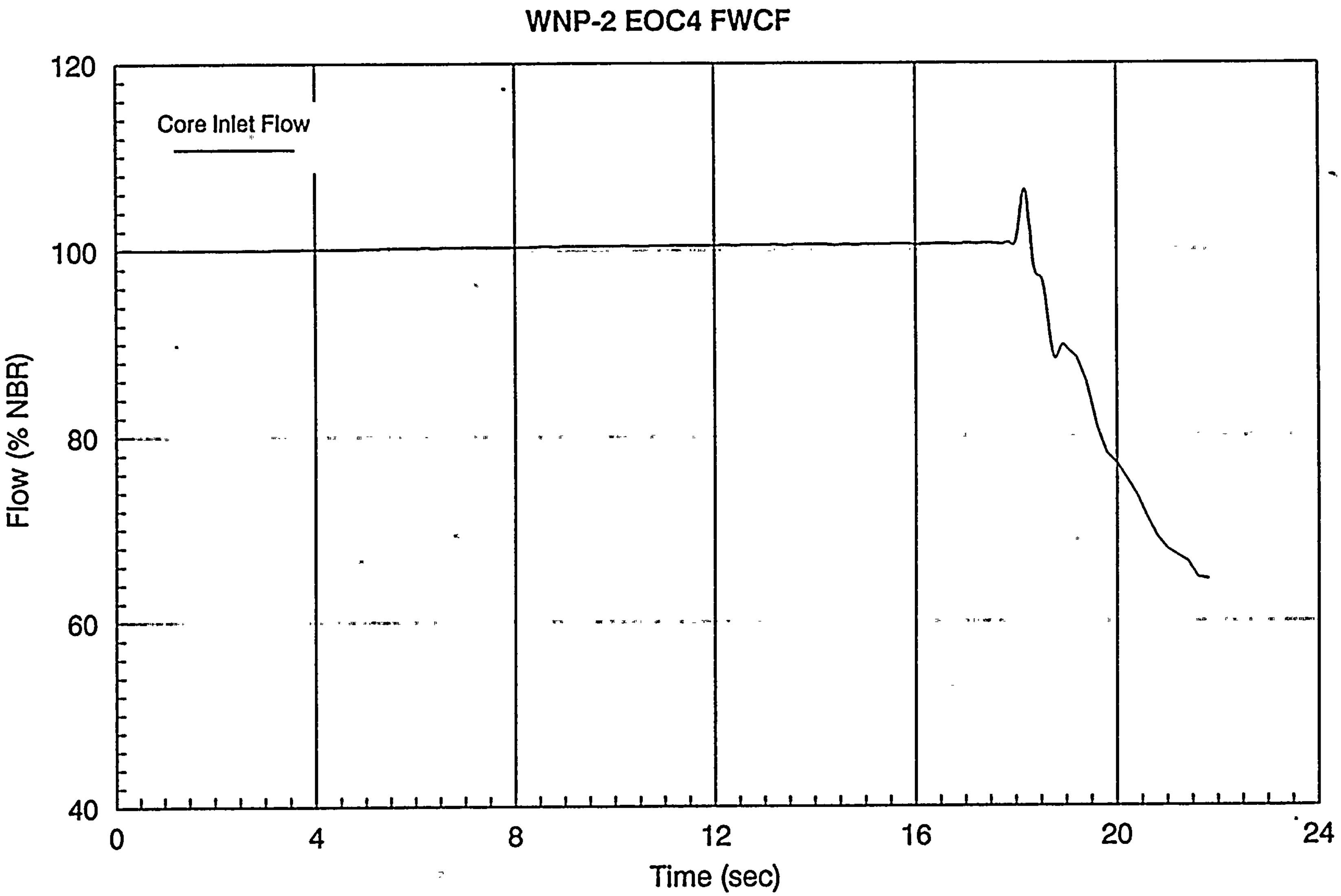

Figure 5.3.7-10 FWCF Results, RPT Operable, Tech. Spec. Scram Time

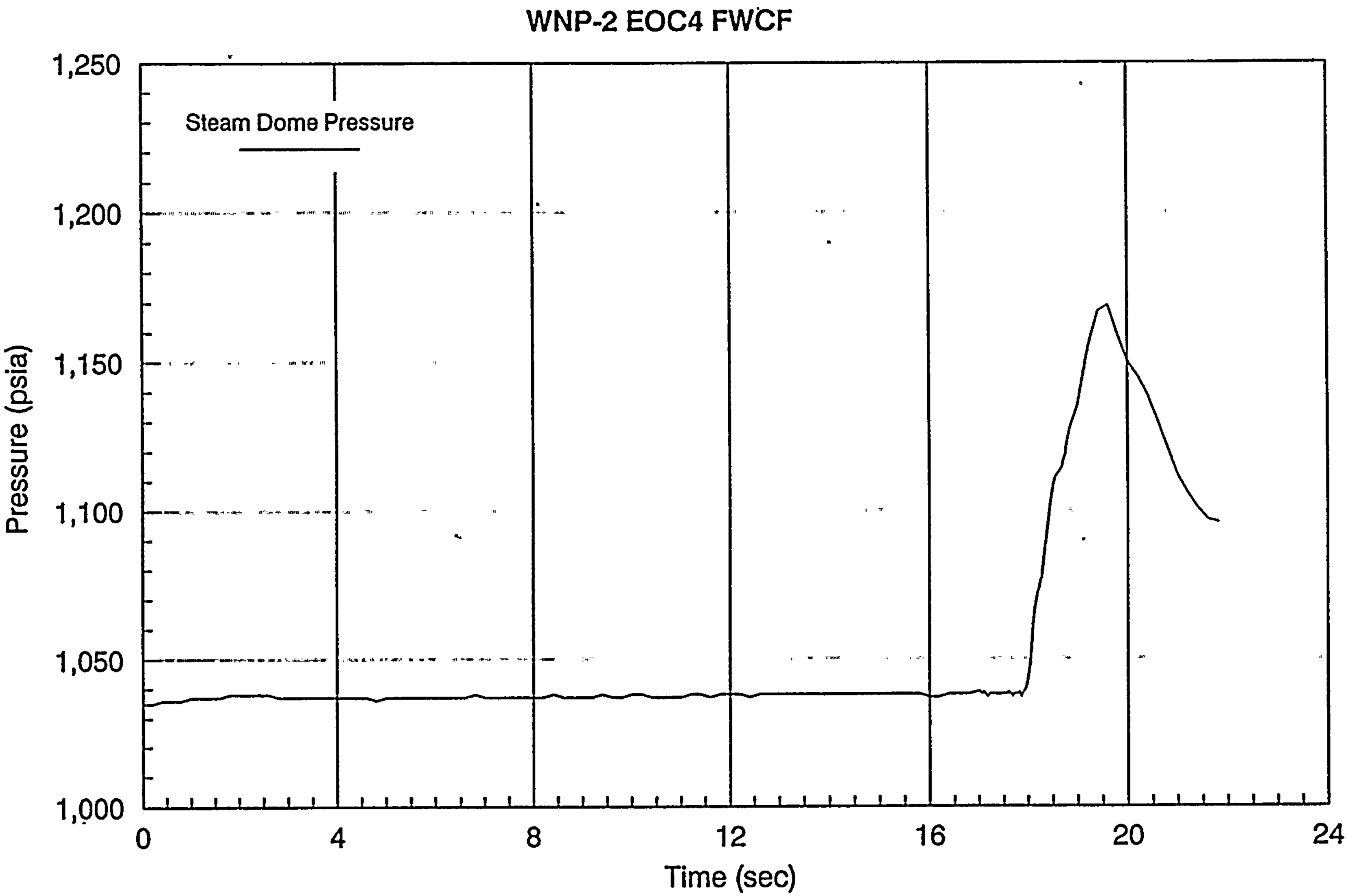

Figure 5.3.7-11 FWCF Results, RPT Operable, Tech. Spec. Scram Time

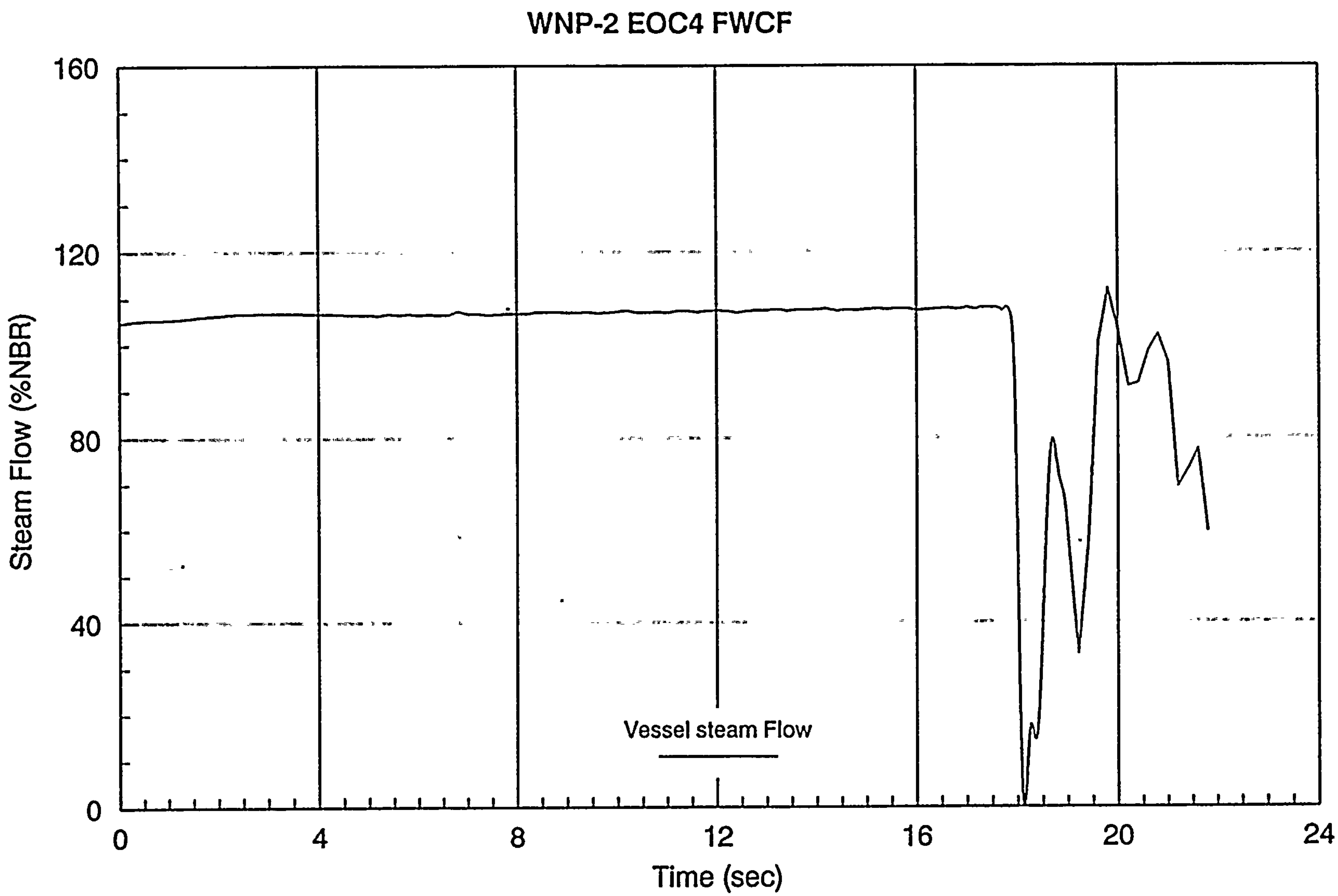

Figure 5.3.7-12 FWCF Results, RPT Operable, Tech. Spec. Scram Time

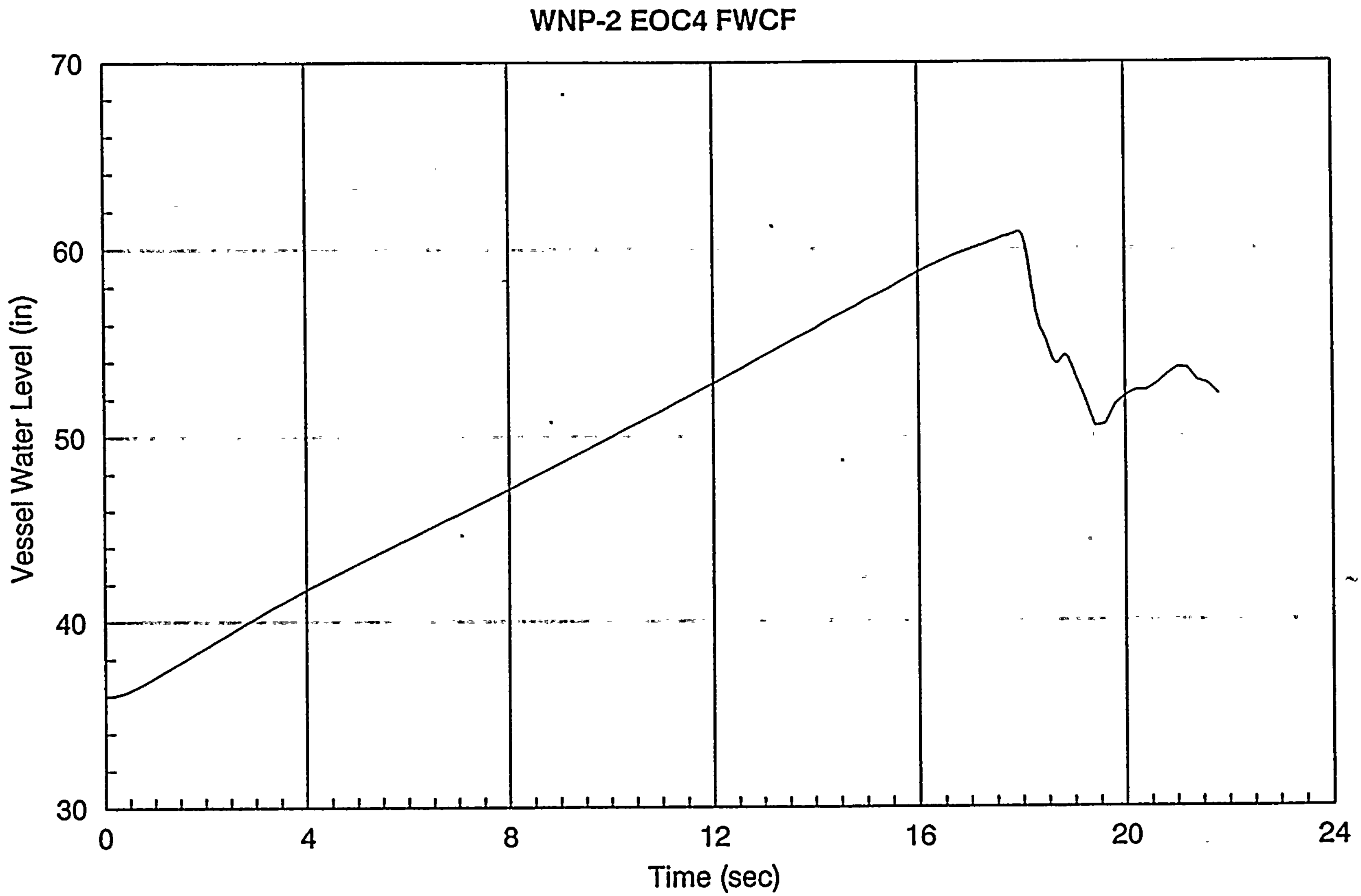

Figure 5.3.7-13 FWCF Results, RPT Operable, Tech. Spec. Scram Time

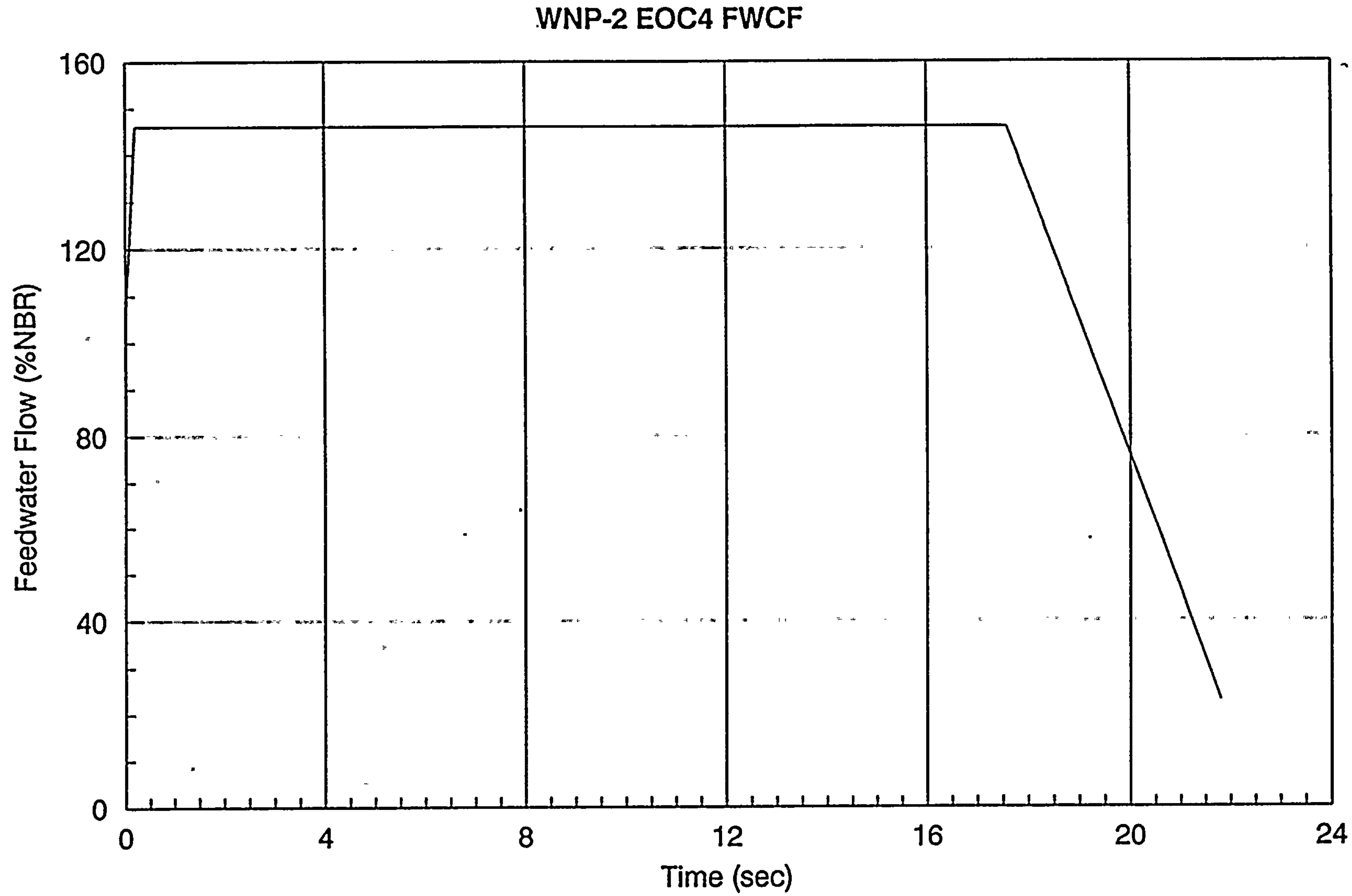

Figure 5.3.7-14 FWCF Results, RPT Operable, Tech. Spec. Scram Time

5.3.8 Recirculation Flow Controller Failure--Increasing Flow

5.3.8.1 Event Description

Failure of the master controller or the neutron flux controller can result in opening of the flow control valves in both recirculation loops. The valve opening speed is limited by system electronics. Also, failure within an individual loop's flow controller can open the flow control valves for that loop. In this case, the valve opening speed is limited by the system hydraulic characteristics. In either of these cases, the increase in recirculation flow causes an increase in core flow, which in turn, results in an increase in core power and shifts the power distribution toward the top of the core. The rate and magnitude of the power increase is dependent on the rate and magnitude of the recirculation flow increase. There are two possible analysis objectives: (1) demonstrate that the most severe recirculation flow increase does not exceed transient limits; and (2) identify reduced flow and power operating limits.

Based on the plant safety analysis, the first case has been shown to be non-limiting with respect to the reload fuel analysis process. Therefore, it is not discussed further with respect to the event analyses required as a part of the reload fuel event analysis process.

It is the second case that is evaluated as a part of the reload analysis process. In this case, the initial conditions and assumptions are chosen to maximize the change in ΔCPR. This is accomplished by the assumption of a gradual flow increase such that the protective actions will not occur prior to the time of operator actions to mitigate the event consequences. This is the case described herein. The results of this analysis can establish the MCPR operating limits at low power and flow conditions.

Table 5.3.8-1 shows the expected sequence of events for the recirculation flow controller failure - increasing flow transient.

5.3.8.1.1 Initial Conditions and Operational Assumptions

The following plant operational conditions and assumptions form the principal bases for the analysis of the recirculation flow controller failure - increasing flow event:

(a) The plant is operating on low power and flow operating limits on the power/flow operating map.

(b) The NSSS operating parameters are consistent with a plant heat balance for the initial power and flow conditions.

(c) A failure in the recirculation flow controller system causes the recirculation flow to slowly increase.

(d) The system trips and initiation signals are consistent with technical specifications.

(e) All of the plant control systems function normally.

5.3.8.1.2 Operator Actions

Operator actions are aimed toward reducing power and controlling recirculation flow. These actions include:

(a) Regain control of the recirculation system flow control valves.

(b) Take the necessary actions to return to a normal operating state.

(c) Identify cause of the failure and take action to initiate repairs.

5.3.8.1.3 Event Acceptance Limits

The acceptance limit for this event is that the MCPR $\geq$ fuel cladding integrity limit. Compliance to the fuel cladding integrity limit is demonstrated by assuring that the operating limit MCPR is greater than or equal to the fuel cladding integrity limit MCPR (which assures that greater than 99.9% of the fuel rods in the core are not expected to experience boiling transition) plus the change in ΔCPR during the event.

5.3.8.2 Analysis Considerations

This section describes the key analysis considerations applicable to the recirculation flow controller failure - increasing flow event. It includes: (1) a description of the phenomena occurring during the event that have a significant impact on the event consequences; (2) a discussion of the system performance characteristics that can significantly affect the course of the event; and (3) a discussion of the performance characteristics of the important components as they relate to the event consequences.

5.3.8.2.1 Key Phenomena

Described below are the key phenomena related to the recirculation flow controller failure - increasing flow event. These phenomena are considered in the simulation of this event in order to accurately model the plant response. The recirculation flow controller failure - increasing flow event is relatively simple event to model because of slow nature of the event.

Recirculation Phenomena Simulation of the recirculation phenomena is important for the analysis of this event. The event begins with a postulated failure that slowly increases recirculation flow, which results in a increased core power and core flow. The flow increase continues until the maximum recirculation flow as limited by the system design, without initiating a scram on high neutron flow or high thermal power, is reached.

Core Phenomena Simulation of the core flow and power increase is important in the analysis of this event. The phenomena important in the core region are the reactivity effects that contribute to changes in the neutron flux and heat transfer to the coolant. During the recirculation flow controller failure - increasing flow transient, the reduction of steam voids due to the increased core flow is responsible for the initial positive reactivity insertion. This causes an increase in power which increases fuel temperature.

Pressure Vessel Phenomena As the recirculation water enters the vessel, it is used to drive the jet pumps which take suction flow from the downcomer region between the core shroud and the vessel wall and discharge to the lower plenum. The flow out of the jet pumps represents the total core flow (except for some minor contribution from the control rod drive cooling flow). The increase in recirculation flow increases the core flow. The water and steam mixture exits from the core to the upper plenum and then through the steam separators and steam dryers.

Steam Line Phenomena As the power level increases, the steam flow will increase and the pressure regulator system will open the turbine control valves, and if necessary the turbine bypass valves, to control turbine inlet pressure.

Feedwater Phenomena The feedwater system is not substantially affected by this transient. Small changes in feedwater temperature are neglected.

5.3.8.2.2 Systems Considerations

The recirculation flow controller failure - increasing flow transient is initiated by an assumed failure in the recirculation flow control system. All other systems are assumed to operate as designed and continue to function throughout the event. No other system or component failures are assumed to occur.

For the limiting event, the important systems to be considered are: (1) the core and fuel system, including the nuclear/thermal-hydraulic coupling; (2) the recirculation system, which limits the maximum flow increase; (3) the steam system, including control valves and stop valves; (4) the feedwater system; and (5) the reactor protection systems average power range monitor (APRM) neutron flux and flow referenced thermal power scram setpoint. For the limiting event, the maximum power level is based on the scram setpoint, which effectively limits the power increase.

5.3.8.2.3 Component Performance Characteristics

Analysis of the slow flow increase associated with the recirculation flow controller failure - increasing flow event requires modeling of the core and fuel only. The performance of the other components is associated with the selection on the initial and final operating states.

5.3.8.3 Methodology/Integration of Codes and Analysis

The primary analysis model in the simulation of the recirculation flow controller - increasing flow event is the core thermal hydraulic model, VIPRE-01. VIPRE-01 is used to calculate the changes in core thermal hydraulic parameters based on the assumed changes in core flow and core power during the event. The VIPRE-01 analysis results are used to determine the ΔCPR from the operating statepoints evaluated.

The analysis of the recirculation flow controller failure - increasing flow is performed using the following codes in the sequence shown on Figure 2-1: (1) ESCORE; (2) MICBURN-E; (3) CASMO-2E; (4) FICE; and (5) VIPRE-01. ESCORE is used to provide the fuel rod temperature distribution used in CASMO-2E and the gap conductance used in VIPRE-01. MICBURN-E provides the gadolinia cross sections used in CASMO-2E. CASMO-2E is used to perform the lattice physics analysis to generate the local peaking factors used by FICE. VIPRE-01 is used to determine the ΔCPR during the transient based on the local peaking function provided by FICE.

5.3.8.4 Licensing Analysis Procedure

The analysis of the recirculation flow controller failure - increasing flow event follows a six step procedure:

(1) Determine the discrete power and flow points to be analyzed. The calculations assume that the event is initiated from the 104% rod line at minimum flow and terminates at 120% power and 103% flow (flow control valve wide open). The power and flow discrete points along this line are then calculated.

(2) Determine core pressure as a function of core power.

(3) Perform a heat balance calculation to determine steam flow, feedwater flow, and feedwater enthalpy for the range of power and flow conditions. Include the recirculation pump power.

(4) Perform heat balance calculations to determine core inlet enthalpy for the range of power and flow conditions.

(5) Calculate core bypass flow. Determine active core flow from total core flow and bypass flow.

(6) A radial power distribution that puts the high power fuel assembly on the safety limit MCPR at 120% power and 103% flow is chosen. Calculate MCPR along the power/flow line that was determined in step 1. This calculation is performed by a steady-state VIPRE-01 analysis. The VIPRE-01 boundary conditions of pressure, inlet enthalpy, and inlet flow are determined in Steps (2), (4), and (5), respectively.

5.3.8.5 Sensitivity Studies/Justification of Procedure

The recirculation flow controller - increasing flow event is based on a conservative determination of the ΔCPR between various reduced power and flow operating states and the maximum core flow and power level that can be attained as a result of a slow increase in recirculation flow. The analysis assumes that the event is initiated from the control rod line on the power flow map associated with 105% steam flow (104.2% power). The event is assumed to be terminated at 120% power and 103% core flow. The core flow at the 120% power level based on the maximum that can be attained with the recirculation flow system design.

The power flow relationship is bounding for a slow increase in core flow assuming no increase in xenon concentration in the fuel during the event (constant xenon assumption). The constant xenon assumption is more limiting than the assumption of xenon buildup during the event. To determine the conservatism in the operating states evaluated for the event, the analysis of changes in operating state assuming constant xenon between the initial operation state and the maximum (with flow control valve wide open) core flow was made using the SIMULATE-E code. For this case, the final power level obtained was 112.5%. This analysis demonstrates

the conservatism in the selection of the operating states used in the analysis of this event.

5.3.8.6 Typical Results

The licensing analysis procedure for evaluating the recirculation flow controller failure - increasing flow event is described in Section 5.3.8.4. This procedure has been used to analyze the recirculation flow controller failure - increasing flow event for Cycle 4. The analysis results are provided in Table 5.3.8-2. The results of the fuel supplier (ANF) are also shown on Table 5.3.8.-2. The primary difference in the analysis is that the Supply System results are based on the use of the approved ANFB critical power correlation, and the ANF results are based on the use of the XN-3 critical power correlation. Therefore, it is concluded that the Supply System licensing analysis procedure is reasonable based on the analysis conservatisms and the comparison to the ANF results.

Table 5.3.8-1

Sequence of Events for
Recirculation Flow Controller Failure - Increasing Flow

Recirculation flow controller fails and recirculation flow begins to increase.

Turbine control valves, and if necessary bypass valves, open to control reactor pressure.

New steady state power level is reached at maximum recirculation flow.

Table 5.3.8-2

Reduced Flow Operating Limit Based on Recirculation Flow Controller - Increasing Flow

Core Flow (% Rated)	Supply System Reduced Flow MCPR Operating Limit	Fuel Supplier Reduced Flow MCPR Operating Limit
103	1.06	1.06
90	1.14	1.12
80	1.22	1.17
70	1.30	1.24
60	1.38	1.32
50	1.46	1.42
40	1.54	1.55

5.3.9 Control Rod Drop Accident

5.3.9.1 Event Description

There are many ways of inserting reactivity into a BWR; however, most result in a relatively slow rate of reactivity insertion and do not pose a significant challenge to the fission product barriers. It is possible that the rapid removal of a high worth control rod could result in a potentially significant power excursion. The accident that has been selected to bound the consequences of the most rapid reactivity insertion events is the control rod drop accident.

The control rod drop accident is the postulated separation of the control rod blade from the control rod drive, with the blade sticking in the fully inserted position while the drive is withdrawn until a high worth control rod pattern is achieved, followed by the dropping of the blade to the control rod drive position. The dropping of a high worth control rod results in a high local reactivity in a small region of the core and, for large, loosely coupled BWR cores, significant shifts in spatial power generation. The initial rapid power increase is limited by Doppler, void, and moderator reactivity, and final shutdown is achieved by a high neutron flux initiated scram.

For the limiting case assumed in the safety analysis process, fuel failures are predicted to occur as a consequence of this accident.

Table 5.3.9-1 shows the expected sequence of events for the control rod drop accident.

5.3.9.1.1 Initial Conditions and Operational Assumptions

The following plant operational conditions and assumptions form the principal bases for the analysis of the control rod drop accident:

(a) The reactor is critical and the power level is less than 20%.

(b) A control rod blade is separated from its drive and stuck in the fully inserted position.

(c) Control rods are withdrawn to establish a high worth control rod pattern consistent with the constraints of the rod sequence control system (RSCS).

(d) The drive separated from its blade is at a position consistent with the other control rods assigned to its group.

(e) The control rod blade becomes unstuck and drops at its maximum velocity to the position of the drive.

(f) Scram is initiated on high neutron flux.

5.3.9.1.2 Operator Actions

The termination of the excursion is accomplished by the inherent design of the fuel and core and the automatic initiation of a scram. Following the scram, the operator would be expected to confirm that the reactor is shut down and take the necessary actions to regain control of the plant. Assuming the plant is isolated, the operator should take the appropriate actions identified in the plant emergency procedures, which include monitoring reactor water level and pressure and suppression pool temperature.

5.3.9.1.3 Event Acceptance Limits

The acceptance limit for this event is a peak fuel enthalpy deposition of 280 cal/gm. Compliance with the peak fuel enthalpy limit is demonstrated by assuring that the key inputs to the generic analysis will result in a peak radially averaged fuel enthalpy deposition of less than 280 cal/gm.

5.3.9.2 Analysis Considerations

This section describes the key analysis considerations applicable to the control rod drop accident. It includes: (1) a description of the phenomena occurring during the event that have a significant

impact on the event consequences; (2) a discussion of the system performance characteristics that can significantly affect the course of the event; and (3) a discussion of the performance characteristics of the important components as they relate to the event consequences.

5.3.9.2.1 Key Phenomena

Described below are the key phenomena related to the control rod drop accident. Consideration of these phenomena is necessary in the simulation of this event to accurately model the plant response.

The major complexity in analyzing this event is in the simulation of the core phenomena, including the changes in the core power and power distribution during the initial power excursion. From a systems standpoint, the control rod drop accident is relatively simple to model because the majority of the system actions occur after the challenge to the accident acceptance limits is terminated. The primary systems considerations are in the simulation of the normal operating systems as they limit control rod worth and the performance of the control rod drive system as it inserts scram reactivity. These phenomena are considered as part of the core phenomena. Other system phenomena are associated with evaluating the radiological consequences of calculated fuel failures.

Core Phenomena The event begins with the dropping of a fully inserted high worth control rod that had been previously separated from its drive. The control rod blade is then assumed to drop at its maximum velocity to the position of its drive. The maximum control rod worth is constrained by the normal operating systems as they limit the allowable control rod patterns. The maximum velocity of the falling control rod is limited by the design of the control rod blade.

The postulated dropping of a control rod results in a high local reactivity addition in a small region of the core, a substantial local power increase, and a shift in power distribution during the course of the event. The phenomena important in the core region are the reactivity effects that contribute to the changes in the local neutron flux and the energy deposition in the fuel rods. The primary effects are the reactivity changes due to the dropped control rod, the feedback effects of Doppler, void and moderator temperature, and control rod (scram).

The reactivity insertion due to the dropped control rod is dependent on control rod worth, drop velocity, and distance dropped. The rapid change in local power causes a rapid increase in fuel temperature and, depending on the initial reactor conditions, generates additional steam voids or increases moderator temperature. The local power increase is initially limited by the Doppler and void/moderator reactivity feedback and is terminated by reactor scram.

The selection of initial conditions is important in the determination of the reactivity characteristics to be used during the accident. Because the presence of voids tends to mitigate the consequences of the event, zero power startup conditions are limiting. Control rod patterns are selected consistent with these conditions to establish the appropriate accident reactivity characteristics to be used during the postulated accident. The important accident reactivity characteristics are the total control rod worth, accident reactivity shape function, control rod drop velocity, interassembly local power peaking factor, and delayed neutron fraction. The total control rod worth establishes the maximum amount of reactivity that can be inserted as a result of a dropped control rod. The accident reactivity shape function establishes the reactivity inserted as a function of the distance the rod has dropped. The control rod drop velocity establishes the reactivity insertion as a function of time. The maximum interassembly local peaking factor is used to allocate the energy deposition in the fuel assemblies surrounding the control rod. The

delayed neutron fraction is an important parameter in establishing the magnitude of the power increase.

The important reactivity feedback parameters are the Doppler, void, and moderator temperature reactivity. The increase in fuel temperature that results from power increase due to the dropped control rod causes the Doppler reactivity to become more negative, which tends to mitigate the event consequences. In addition, the heat transfer to the moderator from the fuel rods and the direct energy deposition into the coolant cause the moderator temperature to increase or generate additional voids, which also provides negative feedback.

The scram reactivity insertion is the mechanism for making the reactor subcritical. The scram reactivity consists of the scram reactivity function and the scram insertion time. The scram reactivity function is the control rod reactivity as a function of the distance that the control rods are inserted. The scram reactivity insertion time is used to establish the scram reactivity as a function of time.

<u>Systems Phenomena</u> Because the evaluation of this accident is performed with a core dynamics code, the external system phenomena are treated through the use of code inputs. The primary systems inputs are those which impact core reactivity. These include the normal system operating constraints on control rod patterns and the control rod drive (scram) system and are discussed above. The remaining system inputs are treated as steady state because the event is extremely rapid and the system controls do not respond in a manner to significantly effect the course of the event.

5.3.9.2.2 Systems Considerations

The control rod drop accident is the postulated separation of a control rod from its drive, the sticking of the blade in the fully inserted position, the withdrawal of control rods to establish a high worth pattern, and the subsequent dropping of the blade to the

position of the drive. The important systems to be considered in the accident analysis are: (1) the core and fuel system; (2) the RSCS; and (3) the reactor protection and control rod drive (scram) systems, including the high neutron flux initiated scram and the rapid insertion of the control rods. Other systems that may be initiated as a result of an isolation or the decision to change the operating state as part of the accident recovery process are not required because their action occurs after the challenge to the event acceptance limits has occurred. With the exception of the RSCS, which places constraints on the allowable control rod pattern, the system components as discussed in Section 5.3.9.2.3 limit the system performance characteristics.

For WNP-2, the RSCS coupled with the banked position withdrawal sequence constrains the control rod pattern during plant startup and low power operation. The RSCS is a system that constrains control rod movements to predetermined patterns and consequences by initiating control rod blocks to assure that allowable control rod withdrawal and insertion sequences are followed. The banked position withdrawal sequence limits control rod movements to pre-assigned control rod groups and banked positions, such that the maximum control rod worth is reasonably minimized. The RSCS operates until the power level reaches the setpoint that has been established such that constraints on rod patterns are no longer required for protection against the consequences of the control rod drop accident.

5.3.9.2.1 Component Performance Characteristics

The control rod drop accident requires modeling of the core and fuel systems, of the systems that place constraints on control rod patterns, and of the reactor protection and control rod drive (scram) systems. These systems are modeled by treating them as inputs to the core analysis. The key parameters with respect to the core and fuel systems are discussed with respect to the core phenomena and the constraints on control rod patterns are discussed with respect to the overall systems consideration. This section

describes the analysis inputs that are based on the performance of the key system components.

<u>Control Rod Drop Velocity</u> Each control rod is equipped with a velocity limiter that limits the free fall velocity of the control rod blade that is separated from its drive. The maximum drop velocity determined from experimental test data is used in the analysis.

<u>Neutron Flux Trip Level</u> The average power range monitors (APRM) system high neutron flux level provides the signal that initiates reactor scram to achieve reactor shutdown. The technical specification setpoint for high neutron flux in the power range is used in the analysis, even though other flux trips would occur earlier.

<u>Reactor Protection System Delays</u> The reactor protection system includes the devices to generate signals, conditioning of these signals, and comparison to trip setpoints, and initiates reactor scram. The technical specification response time for the reactor protection system is used in the analysis.

<u>Control Rod Insertion Time</u> The control rod drive system provides the necessary negative reactivity insertion to achieve reactor shutdown. The technical specification for the average control rod insertion time for all control rods at reduced pressure is used in the analysis.

5.3.9.3 Methodology/Integration of Codes and Analysis

The control rod drop accident analysis is performed by integrating the WNP-2 reactor analysis methodology with the fuel supplier control rod drop accident analysis methods. In this approach, the cycle specific analysis input parameters for the reload are used in conjunction with the fuel supplier generic analyses. The following methodology is used in the evaluation of the control rod drop accident for fuel supplied by ANF.

ANF has performed generic parametric evaluations of the consequences of the control rod drop accident to determine the maximum enthalpy deposition based for a range of values for the control rod worth of the dropped control rod, four-assembly local power peaking factors, Doppler coefficient, and delayed neutron fraction [Reference 21]. For conservatism, the ANF parametric studies neglect the contributions of void reactivity, so the void coefficient does not appear in their analysis. In addition, the ANF generic parametric analysis utilizes a conservatively low scram reactivity insertion rate and a conservatively high reactivity insertion rate for the dropped control rod.

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 ANF generic control rod drop accident analysis, to obtain a conservative bound for the fuel rod enthalpy deposition. The following codes are used in the sequence shown on Figure 2-1 to determine the input parameters for the generic parametric analysis: (1) MICBURN-E; (2) CASMO-2E; (3) NORGE-B; (4) SIMULATE-E; and (5) SIMTRAN-E. MICBURN-E provides the gadolinia cross sections used in CASMO-2E. CASMO-2E is used to perform the lattice physics analysis to generate the cross sections for SIMULATE-E, the total effective delayed neutron yield for SIMTRAN-E, the Doppler coefficient, and in conjunction with SIMULATE-E, the four-assembly local power peaking factor. NORGE-B is used to transfer the CASMO-2E data to SIMULATE-E and SIMTRAN-E. SIMULATE-E is used to determine the control rod worth and, in conjunction with CASMO-2E, the four-assembly local power peaking factor. SIMTRAN-E is used to develop a conservative value for the delayed neutron fraction.

5.3.9.4 Licensing Analysis Procedure

In the analysis of the control rod drop accident, the following analysis assumptions are made:

(a) The plant is at hot zero power conditions.

(b) Moderator and fuel temperatures are equal.

(c) The coolant void fraction is 0%.

(d) The core is in a xenon free state.

(e) The control rod patterns are consistent with the banked position withdrawal sequence and the RSCS is operable.

(f) Up to eight control rods may be inoperable and are assumed to be fully inserted, with the inoperable control rods separated by at least two operable control rods in all directions.

(g) No more than three inoperable control rods exist in any one RSCS rod group.

The evaluation of each of the four parameters needed for use in the ANF generic parametric analysis is discussed in more detail below.

The maximum worth control rods are identified by core analysis calculations. Candidate high worth rods are identified consistent with the rod withdrawal sequence for the operating cycle being evaluated. The inoperable control rods all are assumed to lie in one half of the core and rods in the other half are investigated as high worth candidates. This assumption conservatively increases the predicted worth of candidate rods. For each candidate rod, the rod worth is determined by a pair of core calculations, one with the candidate rod fully inserted, the other with the rod fully withdrawn. In this way, the top few highest worth rods are identified and, after the remaining three parameters are evaluated, the enthalpy deposition in the adjacent bundles is calculated as described below. This is done to ensure that the highest enthalpy deposition, not just the highest rod worth, is obtained.

The four-assembly local peaking factors are calculated for each of the highest worth rods. The four-assembly local power peaking factor is evaluated by combining the core predictions of assembly power with the lattice physics analysis values of assembly local peaking factors.

The Doppler coefficient is evaluated for each fuel type from pairs of lattice physics calculations at different fuel temperatures. The core average Doppler coefficient is evaluated by taking a core average of these results weighted by fuel type and number of bundles.

The delayed neutron fraction is evaluated by condensation of three-dimensional core data to a core average value. The delayed neutron fraction is based on the lattice physics data.

For each of the high control rod worth rods identified above, the values of rod worth, four-assembly local power peaking factors, Doppler coefficient, and delayed neutron fraction are used along with the ANF generic control rod drop parametric results to evaluate the associated maximum fuel rod enthalpy deposition. The results for each high worth rod are then examined to identify the largest value and to verify that it does not exceed 280 cal/gm.

It is assumed that fuel clad failures may occur in all fuel rods in which the CRDA radially averaged enthalpy deposition exceeds 170 cal/gm at any axial location. In all WNP-2 cycles to date the predicted CRDA peak enthalpy deposition has been considerably less than 170 cal/gm. To prepare for the eventuality that future changes in cycle length or fuel design might lead to predicted peak CRDA enthalpy depositions exceeding 170 cal/gm, the Supply System has developed a procedure to analyze that situation. The nodal power shape predicted by the core simulation code during the rod worth calculation is combined with the pin local peaking factors from the lattice physics code and the maximum enthalpy deposition from the vendor's generic analysis to estimate the number of fuel whose maximum enthalpy deposition exceeds 170 cal/gm. The radiological consequences of the failure of all such fuel rods is then compared to the scenario analyzed in the WNP-2 FSAR (the failure of 770 GE 8x8 fuel rods) to show that the FSAR analysis is bounding. If the FSAR analysis were not bounding, then new radiological releases would be determined and evaluated.

5.3.9.5 Sensitivity Studies/Justification of Procedure

The control rod drop accident analysis is based on the use of approved fuel supplier methodology. The Supply System inputs to generic parametric analysis are conservatively determined. No further sensitivity studies are required.

Table 5.3.9-1

Sequence of Events for Control Rod Drop Accident

Control rod blade becomes separated from its drive and sticks in the fully-inserted position.

Operator withdraws control rods until the reactor becomes critical with fully-inserted control rod having its maximum worth.

The control rod becomes unstuck and drops at its maximum velocity.

Reactor becomes prompt critical and initial power increase is limited by the Doppler and void/moderator temperature feedback.

Scram occurs on high neutron flux.

Sufficient control rods are inserted to terminate the accident.

5.3.10 Loss of Coolant Accident

5.3.10.1 Event Description

The LOCA is the postulated loss of coolant from the reactor coolant pressure boundary at a rate in excess of the capability of the reactor coolant makeup systems, from pipe breaks up to and including a break equivalent to the double ended rupture of the largest pipe in the reactor coolant system. The analysis requirements for the LOCA are established in 10CFR50.46 and 10CFR50 Appendix K. For BWRs, a LOCA may be postulated for a wide spectrum of break locations and break sizes. Responses to the postulated break may vary significantly over the break spectrum. The largest break is the postulated double ended rupture of a recirculation pipe; however, the largest break is not necessarily the most severe challenge to the performance of the ECCS. This leads to the requirements to evaluate the entire break spectrum and a number of possible single failures.

Regardless of the initiating break characteristics, the response of the plant to a LOCA can be separated into three phases: (1) the blowdown phase; (2) the refill phase; and (3) the reflood phase. During the blowdown phase, there is a net loss of coolant inventory, an increase in fuel cladding temperature, and uncovery of the core. In the refill phase, the ECCS is functioning and there is a net increase in coolant inventory, and the heat transfer to the coolant is less than the decay heat rate of the fuel, resulting in a continued increase in the fuel cladding temperature. In the reflood phase, the coolant inventory has increased to the point where the core recovers, and the fuel cladding temperature decreases. The relative duration of each phase is dependent on the break size, location, and available ECCS components.

The LOCA scenarios vary considerably over the spectrum of break sizes, locations, and equipment failure combinations. For analysis purposes, the limiting LOCA has been demonstrated in the safety analysis to be a large break in the recirculation system suction

line with the assumed failure of the diesel generator that power the low pressure core spray. This is the event described below.

Following the postulated break, reactor vessel pressure and core flow begin to decrease. The initial pressure response is dependent on the closure characteristics of the main steam line isolation valves and the relative values of the energy added to the system by the decay heat and the energy removed from the system by the initial blowdown of the fluid from the downcomer region of the reactor pressure vessel. The initial core flow decrease is rapid because the recirculation pump in the broken loop loses suction almost immediately, and the pump in the intact loop begins to coast down. The pump coastdown controls the core flow for the next several seconds. When sufficient fluid is released from the vessel and the jet pumps uncover, the core flow decrease to near zero. Following recirculation pump suction uncovery, the energy release from the break increases significantly, and the pressure begins to decay more rapidly. During the rapid pressure reduction, the initially subcooled liquid in the lower plenum flashes, increasing core flow for the next several seconds. Water level in the core region remains high during the early stages of blowdown because of the initial fluid inventory and the flashing of the water in the core. After a short time, the level in the core decreases, and the core becomes uncovered. Several seconds later, the ECCS is actuated and the reactor vessel water level begins to increase. The core is rapidly recovered following filling of the lower plenum.

Heat transfer rates on the fuel cladding during the early stages of the blowdown are governed primarily by the core flow response. Nucleate boiling continues in the high powered regions of the core until uncovery of the jet pumps. Boiling transition follows, and film boiling heat transfer rates are applied. Heat transfer rates increase during the period of lower plenum flashing, slowly decreasing as the flashing progresses until the high power region uncovers. Following uncovery, convective heat transfer is assumed to cease, leaving only radiation heat transfer between the fuel

rods and to the channel. Heat transfer following ECCS initiation is based initially on the heat transfer to the core spray, followed by a rapid increase in heat transfer due to core recovery.

The cladding temperature in the high powered region of the core decreases initially because nucleate boiling is maintained and the heat input and fluid temperature decrease. The reactor initially becomes subcritical due to the large increase in core voids, followed by reactor shutdown due to a scram of the control rods. A rapid, short duration cladding heatup follows the time of boiling transition, when film boiling occurs and the cladding temperature approaches the fuel temperature. The subsequent heatup is slower, controlled by the decay heat and core spray heat transfer. Heatup is terminated by core recovery due to the accumulation of ECCS water.

Table 5.3.10-1 shows the expected sequence of events for the LOCA.

5.3.10.1.1 Initial Conditions and Operational Assumptions

The following plant operating conditions and assumption form the principal basis for the analysis of the LOCA.

- o The plant is operating at its maximum power level and at a core flow consistent with the power level.
- o A pipe break occurs at any location in the reactor coolant pressure boundary, any size greater than the capability of the normal makeup system up to and including the double ended break of a recirculation system pipe.
- o The combination of ECCS components operable is the most limiting considering a single failure and a concurrent loss of power, consistent with the definitions in 10CFR50 Appendix A.
- o Only safety related components are available to mitigate the consequences of the LOCA.
- o The ECCS is initiated by high drywell pressure or low reactor water level, which ever occurs first.

- The containment isolation valves are closed by the first isolation setpoint reached.
- The secondary containment is isolated and the standby gas treatment system is initiated by the first setpoint reached.
- No credit is taken for the operation of the reactor core isolation cooling system (RCIC). The RCIC is considered the normal makeup system for the purposes of establishing the minimum break size.

5.3.10.1.2 Operator Actions

Short term actions (scram, ECCS initiation, primary and secondary containment isolation, standby gas treatment system initiation) take place automatically. The operator should take the appropriate actions identified in the plant emergency procedures, which include monitoring reactor water level and pressure and suppression pool temperature. The operator would be expected to confirm reactor shutdown and appropriate system actuations and attempt to initiate operation of any failed components. Upon successful completion of the initial core cooling phase, the operator would be expected to initiate the long term core and containment cooling and begin the accident recovery process.

5.3.10.1.3 Event Acceptance Limits

The event acceptance limits for this accident are contained in 10CFR50.46. These are: (1) a peak cladding temperature $\leq 2200^{\circ}F$; (2) maximum cladding oxidation $\leq 17\%$; (3) core wide metal water reaction $\leq 1\%$; (4) the maintenance of a coolable geometry; and (5) assurance of long term cooling. For the reload analysis process, compliance with the peak cladding temperature and maximum cladding oxidation limits are demonstrated by the ECCS performance analysis for the LOCA, consistent with the 10CFR50 Appendix K model requirements, for the fuel operating at its MAPLHGR. The ECCS performance analysis also demonstrates compliance with the core wide metal water reaction analysis. As described in Section

5.1.4.2 for the reload fuel analysis, compliance with the first three criteria provides assurance that the remaining criteria are satisfied.

5.3.10.2 Analysis Considerations

This section describes the key analysis considerations applicable to the LOCA. It includes: (1) a description of the phenomena occurring during the event that have a significant impact on the event consequences; (2) a discussion of the system performance characteristics that can significantly affect the course of the event; and (3) a discussion of the performance characteristics of the important components as they relate to the event consequences.

5.3.10.2.1 Key Phenomena

Described below are the key phenomena related to LOCA for the limiting break size, consistent with the requirement of 10CFR50.46 and Appendix K. Consideration of these phenomena is necessary in the simulation of this event to accurately model the plant response. The LOCA is a very complex event to model. The primary phenomena to be considered include: (1) blowdown; (2) recirculation; (3) steamline; (4) feedwater; (5) pressure vessel; (6) ECCS; (7) core; and (8) fuel.

<u>Blowdown Phenomena</u> The event is initiated by the assumed instantaneous break in a pipe in the reactor coolant pressure boundary. The fluid at the break entrance is initially subcooled and flashes to a two phase mixture as it passes through the break. Because of the initial pressure reduction, bulk flashing will occur in the downcomer region and two phase flow continues until the decrease in water level in the downcomer region of the pressure vessel leads to recirculation suction line uncovery. Following uncovery of the suction line, the blowdown flow is primarily steam with entrained liquid.

Recirculation Phenomena The recirculation flow during the blowdown phase is important because of its effect on core flow. The unbroken recirculation loop will experience a coastdown due to the loss of power to the recirculation system pumps. The broken loop will experience reverse flow through the pump during coastdown. The core flow is dependent on the recirculation pump and jet pump characteristics, including consideration of the fluid characteristics during blowdown.

Steamline Phenomena The main steam lines will be isolated as a result of the inventory loss during blowdown. Closure of the main steam line isolation valves (MSIVs) will reduce the depressurization rate.

Feedwater Phenomena Feedwater flow is assumed to coastdown rapidly during the event.

Pressure Vessel Phenomena The most important pressure vessel phenomena are those that influence the coolant inventory and its distribution, including the effect on water level. Water level in the downcomer region is important because it determines the time of jet pump and recirculation suction uncovery during the blowdown phase. Jet pump uncovery significantly affects the core flow, and recirculation system suction uncovery significantly affects the reactor vessel depressurization rate. Because the structural stored energy is significant, heat transfer from the internals to the coolant is included in the analysis.

ECCS Phenomena The ECCS are assumed to initiate at the appropriate setpoints. The limiting single failure is the low pressure core spray diesel generator. The HPCS is initiated consistent with the system startup time. The low pressure coolant injection begins as soon as the reactor vessel pressure reaches the pressure permissive setpoint and the injection valves open.

Core Phenomena The important core phenomena during the LOCA include the core inlet flow, flow between the core bypass and the

fuel assemblies, and counter current flow limitations at the top and bottom of the core.

Initially, the core flow rapidly coasts down consistent with the characteristics of the unbroken loop and essentially stops at the time of jet pump uncovery. When the system pressure falls below the saturation pressure of the lower plenum, lower plenum flashing occurs and significant core flow rate is reestablished. Following lower plenum flashing, the core flow stops until startup of the HPCS system begins spraying water into the core region. The core is reflooded by the combined action of the core spray system and low pressure coolant injection pumps.

One of the most important phenomena during the refill phase of the LOCA is counter current flow limitations. Counter current flow limitations occur when the vapor flow through a restricted area is sufficiently high to impede the liquid drainage. Counter current flow limitations can delay the reflooding of the core and termination of the core heatup process. There are three areas of interest with respect to counter current flow limitations during a BWR LOCA: (1) at the upper tieplate of the fuel assembly; (2) at the top of the core bypass due to the top fuel guide; and (3) at the fuel assembly inlet orifice.

During the refill phase, the bypass region will refill faster than the fuel assemblies. Flow through the bypass flow paths aids in reflooding the vessel and fuel assemblies. Also the water in the bypass region assures that the channel remains a primary heat sink for the fuel rods.

<u>Fuel Phenomena</u> The primary fuel phenomena of interest during a LOCA include the internal heat generation, conduction heat transfer within the fuel pellets, fuel rod pressure, and cladding swelling and rupture, chemical reaction between the cladding and coolant, and radiant heat transfer between the fuel rods and to the channel wall. Internal heat generation, along with the stored energy in the fuel, is the primary source of energy for increasing fuel

cladding temperature during the LOCA. Internal heat generation is due to delayed neutron induced fissions, decay of the actinides, and decay of the fission products. Conduction heat transfer both within the fuel pellet and in the fuel pellet to cladding gap limits the rate at which the stored energy and the decay heat is transferred to the fuel cladding. During the core heatup phase, the fuel cladding may swell and rupture as a result of the increasing fuel rod internal pressure. At elevated temperature, a significant exothermic hydrogen producing reaction between the cladding and the coolant (metal water reaction) occurs that can significantly increase the predicted cladding temperature. Radiant heat transfer between the fuel rods and to the fuel assembly channel is important in limiting the peak cladding temperature of the fuel assembly.

5.3.10.2.2 Systems Considerations

The LOCA is initiated by the postulated break of a pipe in the reactor coolant pressure boundary. The primary systems associate with the LOCA are the ECCS, which include the HPCS, the low pressure core spray, the low pressure coolant injection, and the automatic depressurization systems. The limiting case is evaluated assuming only the availability of onsite power supplies. In addition, it was determined that the limiting single failure is the failure of the low pressure core spray diesel generator, and it is assumed to be failed for analysis purposes. Other systems associated with the LOCA analysis are the main steam, reactor protection, and control rod drive (scram) systems. The system performance characteristics are limited by the key components in the system as described in Section 5.3.10.2.3.

5.3.10.2.3 Component Performance Characteristics

For the LOCA, detailed modeling of the reactor vessel, core, and fuel is required. The ECCS is required to provide core cooling; the reactor protection system and control rod drive (scram) system

is required to achieve reactor shutdown; and the main steam system impacts the vessel blowdown. The key components within these systems limit their performance. The selection of the licensing basis component performance characteristics is based on a buildup of conservative assumptions established by past practices and licensing requirements.

<u>ECCS Initiation Delays</u> The systems initiating the ECCS are comprised of a number of analog and digital signals. Time delays are based on the maximum values contained in the technical specifications.

<u>ECCS Flow Rates</u> The minimum pump flow characteristics based on the technical specification flow rates are used.

<u>ECCS Injection Valve Opening Times</u> The slowest ECCS injection valve opening times identified in the technical specifications are used.

<u>MSIV Closure Time</u> The slow rate of MSIV closure time identified in the technical specification is used to maximize the inventory loss.

<u>Reactor Protection System Signal Delay</u> The reactor protection system includes the devices to generate signals, conditioning of these signals, and comparison to trip setpoints, and initiates reactor scram. The technical specification response time for the reactor protection system is used in the analysis.

<u>Control Rod Insertion Time</u> The control rod drive system provides the necessary negative reactivity insertion to achieve reactor shutdown. The technical specification for the average control rod insertion time for all control rods at reduced pressure is used in the analysis.

5.3.10.3 Methodology/Integration of Codes and Analysis

The LOCA analysis is performed by integrating the WNP-2 reactor analysis methodology with the fuel supplier LOCA analysis methods. In this approach, the cycle specific analysis input parameters for the reload developed by the Supply System are used by the fuel supplier in the performance of the LOCA analysis. The following methodology is used in the development of the LOCA input parameters for fuel supplied by ANF. The approved ANF LOCA analysis methodology is documented in Reference 10.

For each operating cycle, the parameters evaluated by the Supply System are the local peaking factors for each fuel type, the fuel rod power histories, the void reactivity coefficient, the Doppler coefficient, and the time dependent scram reactivity. The following codes are used in the sequence shown on Figure 2-1 to determine the input parameters for the generic parametric analysis: (1) MICBURN-E; (2) CASMO-2E; (3) NORGE-B; (4) SIMULATE-E; (5) SIMTRAN-E; and (6) RETRAN-02. MICBURN-E provides the gadolinia cross sections used in CASMO-2E. CASMO-2E is used to perform the lattice physics analysis to generate the cross sections for SIMULATE-E, the total effective delayed neutron yield for SIMTRAN-E, the Doppler coefficient, and the local peaking factors. NORGE-B is used to transfer the CASMO-2E data to SIMULATE-E and SIMTRAN-E. SIMULATE-E is used to determine the void reactivity and to provide the three-dimensional core data to SIMTRAN-E. SIMTRAN-E is used to develop the delayed neutron fraction and provide input to RETRAN-02 for the determination of the scram reactivity. Separate RETRAN-02 cases are run to determine the scram reactivity.

5.3.10.4 Licensing Analysis Procedure

In the analysis of the limiting break size associated with the LOCA, the following analysis assumptions are applied.

(a) The plant is operating at a power level of at least 1.02 times the licensed power level and up to the maximum core flow. The MAPLHGR at any plane in the limiting fuel

assembly is 1.02 times the operating limit. The power level and MAPLHGR requirements are consistent with the requirements of 10CFR50 Appendix K.

(b) A large pipe break occurs in the recirculation suction line.

(c) There is a simultaneous loss of offsite power.

(d) The low pressure core spray diesel generator fails to start.

(e) Only safety related components are available to mitigate the consequences of the LOCA.

(f) The ECCS is initiated by the first setpoint reached.

(g) The containment isolation valves are closed by the first isolation setpoint reached.

(h) The secondary containment is isolated and the standby gas treatment system is initiated by the first setpoint reached.

(i) No credit is taken for the operation of the reactor core isolation cooling system (RCIC).

The parameters evaluated by the Supply System for use by ANF in the LOCA analysis process are the local peaking factors for each fuel type, the fuel rod power histories, the void reactivity coefficient, the Doppler coefficient, and the time dependent scram reactivity. The first two quantities (local peaking factors and fuel rod power histories) are transmitted to ANF and are used to evaluate the peak clad temperature and the initial fuel rod stored energy. For these evaluations, ANF uses the approved analysis methods and verifies that the peak clad temperature is less than 2200 °F and that the initial fuel rod stored energy lies within the range considered in the LOCA analysis. The remaining three quantities (void reactivity, Doppler coefficient, and scram reactivity) are compared by the Supply System to the values used in the ANF generic analysis to insure that the ANF values provide conservative bounds for the consequences of a LOCA.

The evaluation of each of the five cycle specific neutronics quantities needed in the LOCA analysis is discussed below.

The local peaking factors are calculated as a function of exposure and voids using the lattice physics methods.

The fuel rod power histories are calculated using the core simulation code. The first step in determining the rod power histories is to do a cycle step through based on the cross section libraries generated by the lattice physics codes. Following these calculations, the fuel rod power histories are determined by combining the nodal power distributions with the assembly local peaking factors.

In evaluating the void reactivity, the all rods out, end of cycle conditions are analyzed because the combination of scram, void, and Doppler reactivities is most limiting under those conditions. The void reactivity is determined as a function of moderator density by running a series of core simulator calculations in which the core pressure and inlet subcooling are changed in combinations that preserve the inlet enthalpy. The resulting reactivities as a function of moderator density are used to verify that they are more negative than those assumed in the LOCA analysis.

The Doppler reactivity is obtained as a function of exposure and fuel temperature from the results of pairs of lattice physics calculations at different fuel temperatures. The results are examined to verify that they are more negative than those assumed in the LOCA analysis.

The scram reactivity is calculated by first condensing the three-dimensional core data for the end of cycle to one-dimensional data by use of the core neutronic linkage code. The resulting one-dimensional data is used as input to the transient analysis code. In transient analysis calculation, the void and Doppler reactivities are set to zero and a control rod scram is modeled in order to determine the scram reactivity. The resulting scram rod worths are compared to the values used in LOCA analysis to verify that the Supply System worths are greater.

5.3.10.5 Sensitivity Studies/Justification of Procedure

The LOCA analysis is based on the use of approved fuel supplier methodology, which satisfies the requirements of 10CFR50.46 and 10CFR50 Appendix K. The required conservatism in the LOCA analysis process is based on the use of 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.

Table 5.3.10-1

Sequence of Events for the Loss of Coolant Accident

A large break in a recirculation pump suction line assumed to occur.

Offsite power is lost.

High drywell pressure and reactor low water level reached.

Scram initiated.

All diesel generators signaled to start. Low pressure core spray diesel generator assumed to fail to start.

ECCS signaled to start.

Reactor low-low water level reached.

MSIVs signaled to close

Reactor low-low-low water level reached.

MSIVs closed.

Diesel generators ready to load and load sequencing begins.

HPCS injection valve open and system reaches design flow.

ECCS low pressure permissive satisfied.

LPCI injection valves open and system reaches design flow.

5.3.11 Shutdown Margin

5.3.11.1 Event Description

A shutdown margin calculation is performed to demonstrate that the core is capable of being made subcritical at all moderator temperatures with adequate margin throughout the operating cycle to account for uncertainties and the effects of a stuck control rod. Shutdown margin is a measure of the degree to which the core is subcritical under the analyzed conditions. The shutdown margin in the cold condition is analyzed for the reference loading pattern and recalculated for the final loading pattern. These analyses provide assurance that the shutdown margin demonstration test required by the plant technical specifications will be acceptable. There is no initiating event associated with the shutdown margin analysis.

5.3.11.1.1 Initial Conditions and Operational Assumptions

The following plant operational conditions and assumptions form the principal bases for the shutdown margin demonstration analysis:

(a) The core is at its most reactive exposure during the operating cycle.

(b) The highest worth control rod is fully withdrawn and all other control rods are fully inserted.

(c) The core is in a xenon free condition.

(d) The moderator temperature is at its most reactive state associated with attaining cold shutdown conditions.

5.3.11.1.2 Operator Actions

There are no operator actions associated with this event because it is an evaluation of plant capability and a demonstration that the plant can attain a safe cold shutdown condition.

5.3.11.1.3 Event Acceptance Limits

The shutdown margin requirement is contained in the plant technical specifications. The criteria used in the analysis process is to assure that there is sufficient margin incorporated to account for analytical uncertainties and assure that the shutdown margin demonstration test will be satisfied. In the core design for the reload fuel analysis to satisfy this criteria, the shutdown margin Δk is required to be greater than 0.01 k_{bcc} (1% of the calculated best estimate cold critical eigenvalue).

5.3.11.2 Analysis Considerations

This section describes the key analysis considerations applicable to the shutdown margin demonstration analysis. It includes: (1) a description of the phenomena occurring during the event that have a significant impact on the event consequences; (2) a discussion of the system performance characteristics that can significantly affect the course of the event; and (3) a discussion of the performance characteristics of the important components as they relate to the event consequences.

5.3.11.2.1 Key Phenomena

<u>Core Phenomena</u> Only the core phenomena are required in the shutdown margin analysis. The systems effects are treated as boundary conditions to the core. The important core phenomena are those that affect the core reactivity in the cold shutdown condition. The reactivity components important in the shutdown margin analysis are: (1) the core reactivity, including exposure effects and reflector representation; (2) the control rod reactivity; and (3) moderator temperature.

5.3.11.2.2 Systems Considerations

The shutdown margin analysis is based on the assumption that the plant is in the cold shutdown condition. All systems are assumed to operate as designed for the cold shutdown mode, with the exception that the highest worth rod is stuck in the fully withdrawn condition.

5.3.11.2.3 Component Performance Characteristics

The shutdown margin analysis requires modeling of the core and fuel system, including control rods, to assure that the reactivity effects are properly accounted for. The remainder of the NSSS provides boundary conditions to the core.

5.3.11.3 Methodology/Integration of Codes and Analysis

The shutdown margin analysis utilizes the SIMULATE-E three-dimensional BWR simulator code as the primary analysis code. SIMULATE-E is used to directly calculate the shutdown margin. The lattice physics input to SIMULATE-E is provided by CASMO-2E through NORGE-B. The MICBURN-E code is used to determine the gadolina cross sections used in CASMO-2E. The RODDK-E code, based on SIMULATE-E input, is used for preliminary identification of candidate high worth control rods. (See Section 2 and Figure 2-1 for an overview of the overall WNP-2 reactor analysis methodology computer code sequence.)

5.3.11.4 Licensing Analysis Procedure

Calculation of the shutdown margin (Δk) requires the use of SIMULATE-E and RODDK-E and associated cold cross section libraries developed using the lattice physics codes. The use of SIMULATE-E and RODDK in the shutdown margin analysis is described below.

Calculation of the shutdown margin requires an extensive set of comparisons of calculated and measured cold critical data to

provide estimates of the computational bias present in the core simulation model. Based on the results of these comparisons, two cold critical correlations have been determined. One of these gives k_{bcc}, a best estimate value of the cold critical eigenvalue (k_{eff}) as a function of exposure. The other gives a conservative value of the cold critical k_{eff} (k_{ccc}), such that for calculated values of k_{eff} below k_{ccc}, there is 95% confidence that the reactor is subcritical.

During the development of a fuel loading pattern for the core, shutdown margin is generally calculated first for a number of preliminary designs and then for the final design. Most details of the shutdown margin calculation are the same for preliminary and final designs. The main steps in the shutdown margin calculation are described below. Figures 5.3.11-1 and 5.3.11-2 are provided as an aid in understanding the shutdown margin calculation.

<u>STEP 1:</u> To begin, a SIMULATE-E hot restart file is generated for each exposure point in the cycle. This is the only step where the details differ for preliminary and final designs:

For a preliminary design, a SIMULATE-E Haling depletion is first run to the end of full power life (EOFPL) with the end of cycle (EOC) exposure selected by the code so the final value of k_{eff} matches a target eigenvalue. The local peaking factors and thermal limits predicted by the code are then checked, and if they are acceptable, a SIMULATE-E reverse Haling depletion is run to generate a hot restart file at each cycle exposure point that is a multiple of 1.0 GWd/MTU from EOC to zero exposure.

For the final design, a set of SIMULATE-E rodded depletions is run starting at the beginning of cycle (BOC) and progressing in exposure steps of 0.5 GWd/MTU. At each exposure step the rod patterns are adjusted until the eigenvalue (k_{eff}) matches a target eigenvalue and the flux shape and thermal limits are acceptable. When these criteria are satisfied the SIMULATE-E hot restart file

is saved and the depletion proceeds to the next exposure point. The process continues until the EOC is reached. By either of the above methods a set of SIMULATE-E hot restart files is generated, one for each exposure point considered.

The remaining steps in the shutdown calculation are performed for each of these exposure points in turn.

STEP 2: At each exposure point, two SIMULATE-E full core, cold calculations are run, one with the control rods in the all rods out (ARO) position and one with the rods in the all rods in (ARI) position. These calculations use the cold cross section library, assume a temperature of 68 °F, and set the xenon concentration to zero. In addition, and for added conservatism, the samarium concentrations are set to their hot equilibrium values, i.e. no reactivity credit is taken for additional samarium produced by promethium decay after shutdown. The two SIMULATE-E runs generate the cold restart files for the ARO and ARI cases.

STEP 3: An initial estimate of rod worths is made to identify likely candidates for highest worth rod (HWR). Because separate SIMULATE-E runs for each rod in the core would require a large amount of computing time and would produce an unwieldy amount of data, this preliminary rod worth evaluation is done with the RODDK-E code. RODDK-E provides a less accurate core model than SIMULATE-E, but requires far less running time. RODDK-E reads the SIMULATE-E cold ARO and ARI restart files, calculates an estimate of the reactivity worth of each control rod, and provides a list of rods (and corresponding eigenvalues) in order of decreasing rod worth.

STEP 4: The RODDK-E results are examined and one or more candidate HWR's are selected for more accurate analysis with the SIMULATE-E code. If RODDK-E identifies a single high worth rod with a corresponding k_{eff} significantly greater than that of the next highest worth rods then it may be appropriate to select only that

rod as the HWR. In most cases though, RODDK-E will identify several high worth rods having very similar values of k_{eff}. Then, all such rods are selected for further study.

STEP 5: A SIMULATE-E full core cold calculation is run for each of the candidate rods selected in the previous step. In each of these calculations, all of the control rods are set to the ARI position except for the associated candidate rod which is fully withdrawn.

STEP 6: The values of k_{eff} obtained by SIMULATE-E in the previous step are examined. The highest eigenvalue is selected and the corresponding withdrawn rod is identified as the true HWR. Because the location of the HWR often changes with exposure this identification of the HWR generally is applicable only to the current exposure point.

STEP 7: In this final step, the value of the shutdown margin (Δk) is determined. This quantity is defined as the difference between the eigenvalue given by kbcc and the SIMULATE-E k_{eff} value (k_{max}) corresponding to the HWR found in the previous step, i.e.:

$$\Delta k = k_{bcc} - k_{max}$$

This completes the calculation of shutdown margin for a particular exposure point. Steps 2 through 7 are repeated for any remaining points.

Shutdown Margin Requirements in Supply System Core Designs

For conservatism, the Supply System requires that the shutdown margin exceed an appropriate minimum value in all situations. The minimum is defined by the simultaneous requirements that:

(a) Δk > (technical specification margin) * k_{bcc} and

(b) $\Delta k > k_{bcc} - k_{ccc}$

The first inequality is a requirement contained in the plant technical specifications. The second inequality is a statement that shutdown margin would remain greater than zero even if it were

defined as $k_{ccc} - k_{max}$ instead of as $k_{bcc} - k_{max}$. This alternative definition would provide a 95% confidence, rather than a best estimate, that the reactor is subcritical. It has been found that the first inequality (a) is always the more demanding. Thus, shutdown margin is required to exceed the technical specification requirement times k_{bcc} in all cases. In core design work, it is current Supply System practice to seek loading patterns having an even more conservative 1% shutdown margin, and in those cases Δk is required to exceed 0.01 times k_{bcc}.

5.3.11.5 Sensitivity Studies/Justification of Procedure

The Supply System requirement that the shutdown margin satisfy (b) above, has built in considerable conservatism because it is equivalent to defining shutdown margin in terms of k_{ccc} (95% confidence) rather than k_{bcc} (best estimate). The use of this more conservative cold critical correlation is believed to bound other uncertainties in the shutdown margin analysis. The correlation k_{ccc} is based on comparisons of measured and calculated cold criticals from 5 cycles at WNP-2. The Supply System will continue to update this correlation as additional cold critical data becomes available.

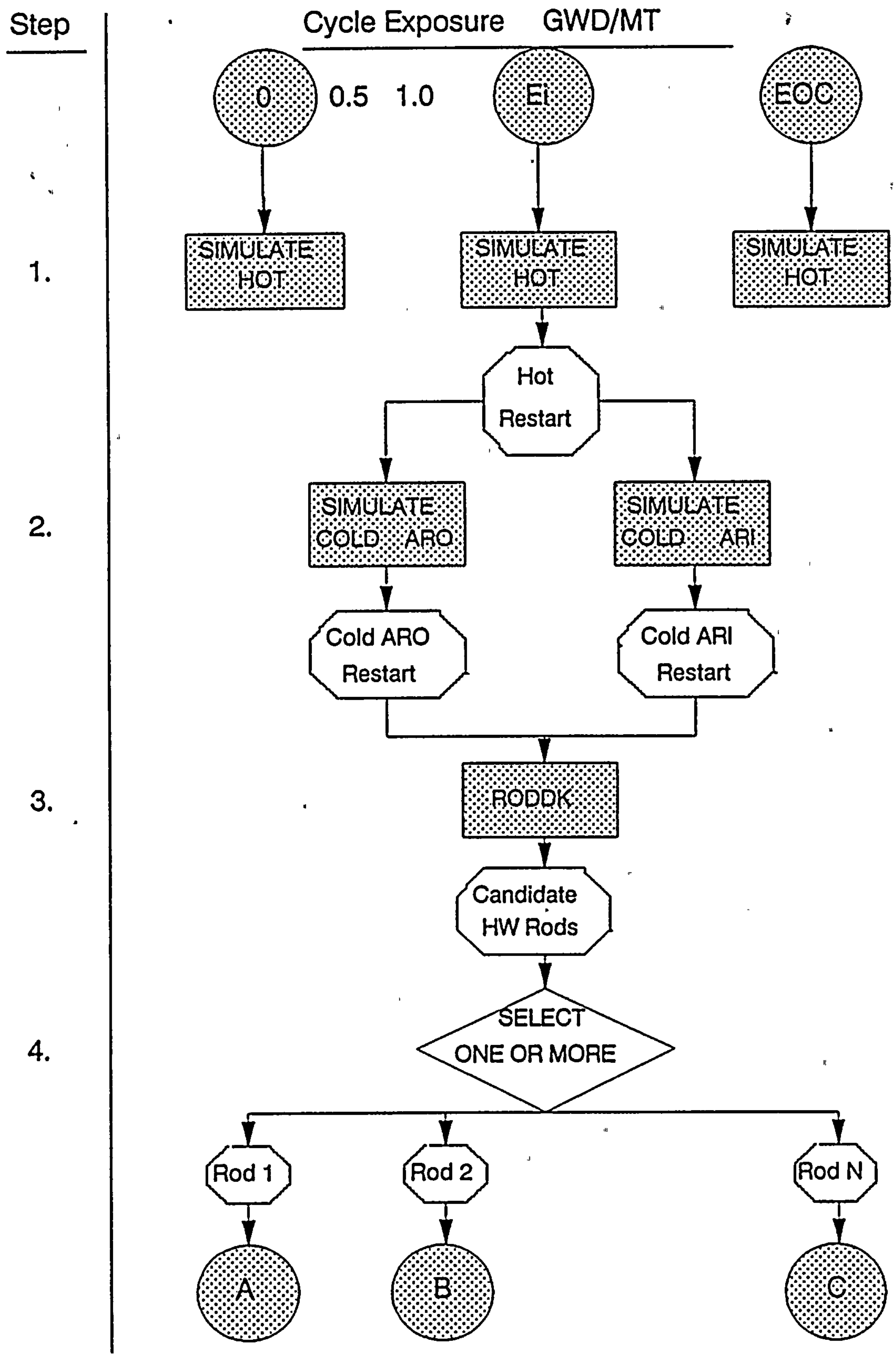

Figure 5.3.11-1

Step

5.

6.

7.

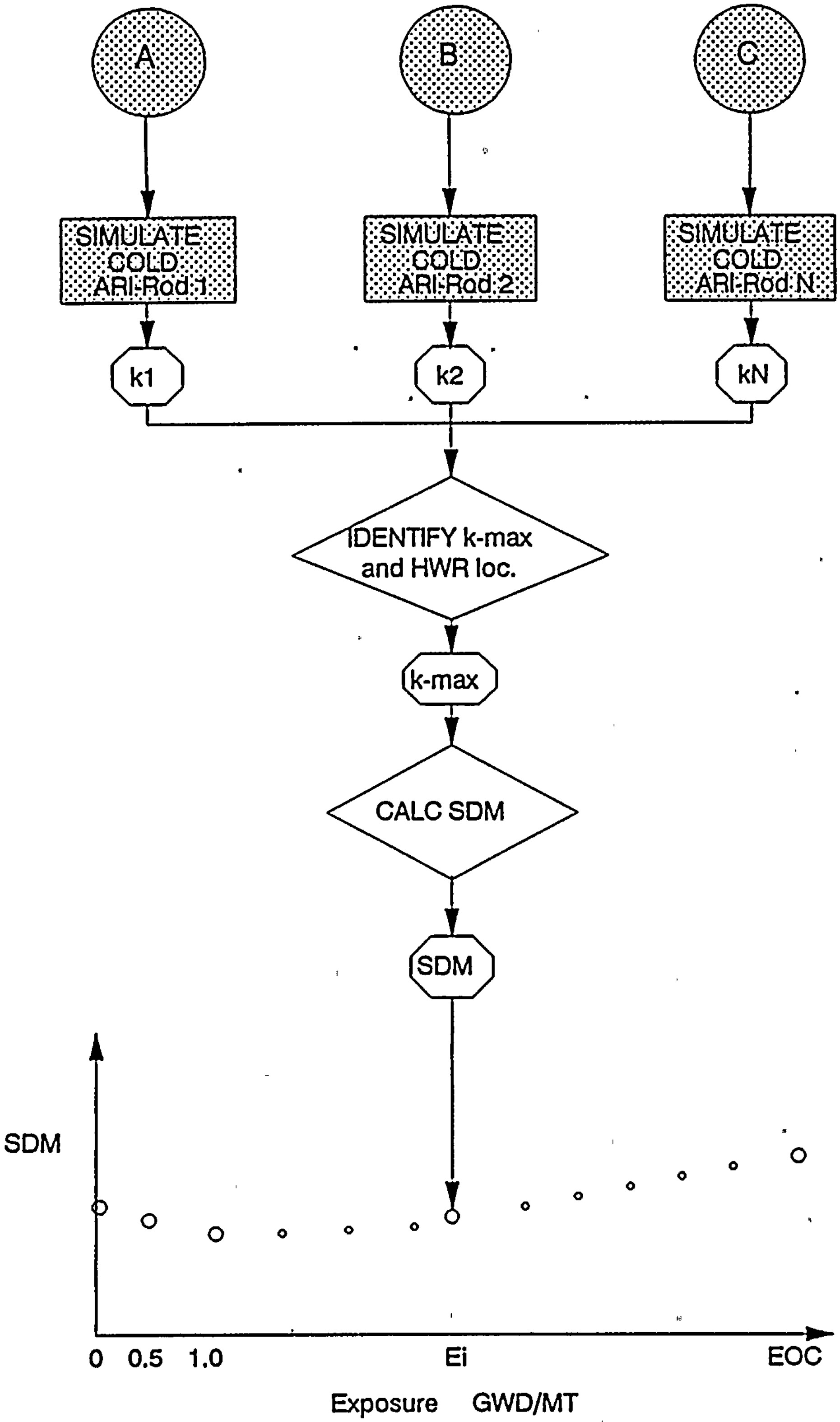

Figure 5.3.11-2

5.3.12 Standby Liquid Control System Capability

5.3.12.1 Event Description

The standby liquid control system capability analysis is performed to demonstrate the ability of the system to perform its design function. The standby liquid control system is designed to insert sufficient negative reactivity to enable the reactor to reach a xenon free shutdown condition from full power operation without movement of the control rods. In the standby liquid control system capability analysis, it is assumed that the control rods remain withdrawn in their full power pattern, and the standby liquid control system is manually initiated to provide the negative reactivity, by injection of sodium pentaborate into the core, necessary to enable the cold shutdown condition to be attained.

Following the manual initiation of the standby liquid control system with the plant initially at full power, the power level will begin to decrease as the sodium pentaborate enters the core. The operator is assumed to follow procedures and control the remaining NSSS parameters such that no automatic system trips or system initiations occur. The event is terminated when the cold shutdown condition is reached. For the purposes of this analysis, shutdown is considered to be accomplished when the reactor is subcritical at the most reactive temperature with no xenon present.

5.3.12.1.1 Initial Conditions and Operational Assumptions

The following plant operational conditions and assumptions form the principal bases for the shutdown margin demonstration analysis:

(a) The plant is at its rated power level and minimum core flow at the most reactive point in the operating cycle.

(b) The control rods remain in a pattern consistent with the operating condition.

(c) The core is in a equilibrium xenon condition.

(d) The standby liquid control system is manually initiated and functions as designed.

(e) The reactor shutdown and subsequent cooldown is controlled by operator actions consistent with plant procedures.

5.3.12.1.2 Operator Actions

Once the operator determines that shutdown is to be accomplished by the standby liquid control system, the system is initiated from the main control room. Following system initiation, system operation and isolation of the reactor cleanup system are confirmed. During the shutdown process, the operator monitors power level and the NSSS operating parameters and follows the procedures for attaining a cold shutdown condition.

5.3.12.1.3 Event Acceptance Limits

The criteria used in the standby liquid control system capability analysis process is to assure that the reactor can attain a cold shutdown condition without movement of the control rods. In the core design for the reload fuel analysis to satisfy this criteria, the shutdown margin is required to be greater than 0.01 Δk at the most reactive temperature with a boron concentration of 660 ppm in the core.

5.3.12.2 Analysis Considerations

This section describes the key analysis considerations applicable to the shutdown margin demonstration analysis. It includes: (1) a description of the phenomena occurring during the event that have a significant impact on the event consequences; (2) a discussion of the system performance characteristics that can significantly affect the course of the event; and (3) a discussion of the performance characteristics of the important components as they relate to the event consequences.

5.3.12.2.1 Key Phenomena

The standby liquid control system capability is a relatively simple event to analyze because the only conditions of importance are the full power operating condition and the shutdown condition at the temperature of maximum reactivity. The major complexity is in the simulation of the core phenomena and the standby liquid control system phenomena as it impacts the core phenomena.

Core Phenomena The core phenomena are the only phenomena that require simulation for the standby liquid control system capability analysis. The phenomena that are of importance in the core region are those that effect the change in reactivity from the full power operating state to the shutdown condition, using sodium pentaborate mixed with the moderator instead of control rods as the mechanism for attaining shutdown.

The negative reactivity insertion by the standby liquid control system must compensate for the reactivity gains in the transition from the full power to the shutdown condition. The positive reactivity effects to be considered include the elimination of steam voids, the reduction in the Doppler reactivity, the reduced neutron leakage from hot to cold, the decrease in control rod worth as the moderator cools, and the decay of the xenon inventory.

The selection of the operating states is important in the analysis. In selecting the full power operating state, the core exposure, control rod pattern, and core flow are selected to maximize the void reactivity and minimize the control rod inventory. In addition, the control rod inventory is established with the maximum xenon inventory (equilibrium xenon). The shutdown condition is established at the most reactive temperature in the range of possible operating conditions between hot shutdown and the minimum temperature within the capability of the shutdown cooling system. The sodium pentaborate solution is simulated as uniformly mixed with a concentration of 660 ppm.

System Phenomena The system effects are treated as steady state inputs to the specific operating state being evaluated. The mixed sodium pentaborate concentration is used to simulate the operation of the standby liquid control system.

5.3.12.2.2 Systems Considerations

The standby liquid control system capability analysis is based on the assumption that the control rods remain in position until the cold shutdown condition is reached. All other systems are assumed to operate as designed.

5.3.12.2.3 Component Performance Characteristics

The standby liquid control system capability analysis requires modeling of the core and fuel system to assure that the reactivity effects are properly taken into account. The remainder of the NSSS including the standby liquid control system is modeled as steady state inputs. The mixed sodium pentaborate concentration is based on an assumed allowance for imperfect mixing, leakage, and the volume of the recirculation system and shutdown cooling system.

5.3.12.3 Methodology/Integration of Codes and Analysis

The standby liquid control system capability analysis utilizes the SIMULATE-E three-dimensional BWR simulator code as the primary analysis code. SIMULATE-E is used to establish the control rod pattern associated with the limiting full power operating condition. This control rod pattern is used at the limiting temperature with the mixed boron concentration to determine the shutdown margin in the xenon free condition. The lattice physics input to SIMULATE-E is provided by CASMO-2E through NORGE-B. The MICBURN-E code is used to determine the gadolinia cross sections used in CASMO-2E and ESCORE provides the fuel temperature distribution in the hot condition. (See Section 2 and Figure 2-1 for an overview of the overall WNP-2 reactor analysis methodology computer code sequence).

5.3.12.4 Licensing Analysis Procedure

Calculation of the shutdown margin (Δk) requires the use of SIMULATE-E associated hot and cold cross section libraries developed using the lattice physics codes, including the boron libraries for the shutdown analysis. In the analysis, a uniform mixture of 660 ppm of boron is assumed in the reactor coolant. Analyses are performed for both the full power at the most reactive exposure point and shutdown condition at the most reactive temperature.

5.3.12.5 Sensitivity Studies/Justification of Procedure

The use of 660 ppm of boron in the core provides the required conservatism in the analysis. The mixing allowance and the allowance for dilution provide adequate assurance that the standby liquid control satisfy its design objective.

Table 5.3.12-1 shows various sensitivity studies performed throughout Cycle 4 and the corresponding margin to criticality for 660 ppm boron in terms of $\Delta k_{subcritical}$. The most reactive exposure is at the beginning-of-cycle 4. The hot operating rod patterns are the control rod patterns at hot full power critical conditions. The rod patterns on the 104% rod line are the control rod patterns developed for the lowest recirculation pump speed allowed at full power which was the 104% rod line. As can be seen the minimum $\Delta k_{subcritical}$ occurs at 560°K at the most reactive exposure in Cycle 4.

Table 5.3.12 -1
Sensitivity Studies

Case	Case Description	Cycle Exposure (GWD/MT)	$\Delta k_{subcritical}$
1	Hot Operating Rod Pattern at 293°K	0.2	0.065
2	Hot Operating Rod Pattern at 425°K	0.2	0.038
3	Hot Operating Rod Pattern at 560°K	0.2	0.017
4	Rod Pattern (104% Rod Line) at 293°K	0.2	0.064
5	Rod Pattern (104% Rod Line) at 425°K	0.2	0.037
6	Rod Pattern (104% Rod Line) at 560°K	0.2	0.016
7	Hot Operating Rod Pattern at 560°K	1.3	0.016
8	Rod Pattern (104% Rod Line) at 293°K	1.3	0.067
9	Rod Pattern (104% Rod Line) at 425°K	1.3	0.038
10	Rod Pattern (104% Rod Line) at 560°K	1.3	0.016
11	Hot Operating Rod Pattern at 293°K	2.8	0.073
12	Hot Operating Rod Pattern at 425°K	2.8	0.042
13	Hot Operating Rod Pattern at 560°K	2.8	0.018
14	All Control Rod out at 293°K	6.0	0.086
15	All Control Rods out at 560°K	6.0	0.027

5.3.13 ASME Code Overpressure Protection Analysis

5.3.13.1 Event Description

The pressure relief system is designed to prevent over-pressurization of the primary reactor coolant pressure boundary. The Code overpressure protection analysis is performed to demonstrate compliance with the ASME Code [Reference 18]. Pressure relief is accomplished by the opening of ASME Code qualified safety/relief valves.

The vessel overpressure protection system for WNP-2 is designed to satisfy the requirement of Section III, "Nuclear Vessels," of the ASME Code. The ASME Code, Section III, permits the reactor pressure vessel and reactor coolant pressure boundary to exceed their design pressures in a pressurization event based on the frequency of occurrence. The code requires that the lowest qualified safety/relief valve setpoint be at or below the vessel design pressure. Section III also allows credit to be taken for the reactor protection system and reactor scram as a pressure protection device when determining the required safety valve capacities for nuclear vessels. The following pressure limits are applied consistent with the ASME Code requirements:

(a) Under upset conditions, the reactor pressure is not to exceed 110% of design pressure.

(b) Under emergency conditions, the reactor pressure is not to exceed 120% of design pressure.

(c) Under faulted conditions, the reactor pressure is not to exceed 150% of design pressure.

Based upon the ASME Code requirements, NRC regulations, and WNP-2 licensing commitments, a conservative approach to the overpressure protection analysis has been adopted. In this approach, the most severe pressurization event is analyzed and the results are compared to the ASME Code upset limits. The event definition includes the failure of any direct scram signal associated with the event initiator. Based on the current safety analysis, the most

severe pressurization event is the closure of all main steam line isolation valves (MSIV) with a high neutron flux scram (i.e., MSIV position switch scram disabled). This conservative approach is used for the WNP-2 analysis. Event occurrence probabilities suggest that it would be more appropriate to compare the ASME emergency or faulted limits to the pressurization consequences of a closure of all MSIVs with a failure of the direct MSIV position switch scram.

Closure of all MSIVs causes a rapid reduction in steam flow which results in a system pressure increase. Neutron flux increases rapidly because of the core moderator void reduction caused by the pressure increase. The pressure increase is limited by opening of the safety/relief valves and the reactor scram that is initiated by the average power range monitor (APRM) high neutron flux signal.

Table 5.3.13-1 shows the expected sequence of events for the Code overpressure protection analysis transient initiated by MSIV closure.

5.3.13.1.1 Initial Conditions and Operational Assumptions

The following plant operational conditions and assumptions form the principal bases for the Code overpressure protection analysis:

(a) The plant is operating at the safety analysis power level and rated core flow.

(b) The remaining NSSS operating parameters are consistent with normal plant operation.

(c) All MSIVs close at the fastest rate identified in the technical specifications.

(d) A reactor scram is initiated only by high neutron flux. No credit is taken for the scram signal generated by MSIV position switches.

(e) The pressure relief function is available to limit the pressure increase. Six safety/relief valves with the lowest setpoints, are assumed to be out of service and

only the safety mode is available for opening of the remaining valves.

(f) All of the remaining plant control systems function normally.

(g) The system trips and initiation signals are consistent with the technical specifications.

5.3.13.1.2 Operator Actions

This event is analyzed to demonstrate compliance to the overpressure protection requirements of the ASME Code. The peak pressure conditions are limited only by systems that initiate automatically. Operator actions are not considered.

5.3.13.1.3 Event Acceptance Limits

The acceptance limits for this event are based on the ASME Pressure Vessel Code, Section III, limits. Compliance with this limit is demonstrating that the peak reactor vessel pressure is less than 1375 psig.

5.3.13.2 Analysis Considerations

This section describes the key analysis considerations applicable to the Code overprotection analysis event. It includes: (1) a description of the phenomena occurring during the event that have a significant impact on the event consequences; (2) a discussion of the system performance characteristics that can significantly affect the course of the event; and (3) a discussion of the performance characteristics of the important components as they relate to the event consequences.

5.3.13.2.1 Key Phenomena

Described below are the key phenomena related to the ASME Code overpressure protection analysis event. Consideration of these

phenomena is necessary in the simulation of this event to accurately model the plant response.

The Code overpressure protection event involves the reactor core, the entire reactor coolant pressure boundary, and the associated valves and discharge piping. The event is characterized by rapidly changing conditions with complex interactions. Because scram is delayed, this is one of the most dynamic events considered in the safety analysis.

Steam Line Phenomena The event begins with the rapid closure of all MSIVs which causes a pressure increase in the steam system that is rapidly transmitted to the reactor pressure vessel by pressure wave phenomena in the steam lines. The relief valves open at pre-established setpoints allowing a steam release path for pressure relief. Nodalization of the steam lines, up to and including the MSIVs, is adequate to assure simulation of the system pressure response. The nodalization of the downstream piping and the turbine control valves is adequate to simulate the control valve and MSIV interactions. Extensive nodalization studies for the steam lines are performed and shown in the attachment to Reference 9.

Pressure Vessel Phenomena The propagation of the pressure wave from the steam lines to the core is an important phenomenon. The attenuation of the pressure wave by the steam separators is a particularly important phenomenon in modeling the timing of the core moderator void changes.

Recirculation Phenomena During the initial phase of the event, the recirculation phenomena are dominated by the rapidly changing conditions in the core and reactor pressure vessel. In the manual flow control mode, the recirculation valve position is held constant. The increase in vessel pressure during the event may be sufficient to initiate a high pressure trip of the recirculation pumps (RPT). Following the RPT, the recirculation pumps will coast down reducing the recirculation flow and core inlet flow.

Modeling of the recirculation flow phenomena is important to assure that the changing recirculation flow is correctly calculated. The recirculation system modeling includes consideration of the downcomer phenomena, including the annular flow region above and through the jet pumps. The changing performance of the jet pumps at varying pressure and drive flow is included. The external recirculation loop flow is represented so that flow into the vessel as a function of time is accurately simulated.

Core Phenomena The phenomena important in the core region are the reactivity effects that contribute to changes in neutron flux level and hence energy generation and power input to the coolant. The primary reactivity feedback effects are steam void reactivity, fuel temperature Doppler reactivity, and control rod reactivity. The steam void reactivity contributes to the initial positive reactivity as a result of steam void collapse as the pressure increase from the steam system reaches the core. The collapsing moderator voids cause an increase in power level which in turn increases fuel temperature and moderator voids to the point that power would reach a new level. With steam flow restricted (unless scram is initiated) pressure and core power level would continue to rise. A rapid scram of control rods limits the magnitude of the transient pressure and power levels.

5.3.13.2.2 Systems Considerations

For the ASME Code Overpressure Protection analysis, the initiating event is the action that causes full closure of all MSIVs. MSIV position scram is not assumed to occur and ASME Code qualified setpoints are used for the safety/relief valves. All other safety systems that respond are assumed to function as designed and to actuate at their pre-established setpoints.

The important systems to be considered are: (1) the reactor protection system including the APRM high neutron flux scram; (2) the control rod drive (scram) system; (3) the steam system

including MSIVs, control valves and safety/relief valves; (4) the recirculation system, including the RPT; (5) the steam separation system inside the vessel; (6) the feedwater system; and (7) the fuel and core system, including the nuclear/thermal hydraulic coupling. Because this analysis is directed toward establishing peak vessel pressure, no other systems are assumed to operate.

5.3.13.2.3 Component Performance Characteristics

The Code overpressure protection analysis requires detailed modeling of the NSSS in order to assure that all systems that influence reactor system pressure, steam flow, core flow, and core inlet enthalpy are properly considered. The selection of licensing basis component performance characteristics is based on a buildup of conservative assumptions established by past practices and licensing requirements.

MSIV Closure Characteristics The main steam isolation valve closing characteristics are fundamental to the analysis because the MSIVs initiate the event by rapidly stopping steam flow from the vessel. The MSIVs are designed to have an adjustable closure time within specified limits and the allowable range is controlled by the technical specifications. Because this event is a pressurization event, the fastest closure time in the technical specifications is used in the analysis to assure a conservative result.

Safety/Relief Valve Setpoints The safety/relief valves are used to protect the reactor coolant pressure boundary against overpressure events. The technical specifications for the ASME Code qualified mode establish limiting conditions for the safety/relief valve setpoints. The maximum values are used in the analysis in order to assure conservative evaluation of the system pressure response. Only the self-actuated (spring) function of the safety/relief valves is provided for in the analysis. The pressure drop on both the inlet and discharge sides of the valves is accounted for.

Further conservatism is obtained by assuming the six SRVs with lowest setpoints are out of service.

Recirculation Pump Coastdown Characteristics The RPT on high reactor pressure is simulated. The slowest pump coastdown consistent with design specifications is used in the analysis. In addition, the maximum instrumentation and signal processing delay times are used in the analysis.

Feedwater Controls The nominal feedwater control response time is used in the analysis. WNP-2 has steam turbine driven feedwater pumps and closure of the MSIVs isolates their steam supply. The coastdown of the feedwater system is simulated following isolation.

APRM Neutron Flux Trip It is assumed that the APRM high neutron flux trip provides the signal to the reactor protection system which, in turn, initiates a reactor scram. This setpoint is controlled by the plant technical specifications and the maximum allowable value is used in the analysis.

Reactor Protection System Signal Delays The reactor protection system includes the collection of a number of analog and digital signals, conditioning of these signals, comparison to pre-established setpoint limits and activation of nuclear system trips. The signal processing and trip initiation involves delay times which impact transient response. The plant technical specifications identify the allowable reactor protection system response times.

Control Rod Drive Insertion Time The control rod drive system provides the primary mechanism for negative reactivity insertion for terminating the transient. The control rod drives are inserted in the scram mode by the scram hydraulic control system. The scram time for the control rods is based on the nominal scram time specified in the technical specifications.

5.3.13.3 Methodology/Integration of Codes and Analysis

The primary model used in the simulation of the event is RETRAN-02. RETRAN-02 is used to calculate the changes in system and core average nuclear and thermal hydraulic parameters throughout the event. The RETRAN-02 analysis results are used in the assessment of peak reactor vessel pressure.

The analysis of the Code overpressure protection analysis is performed using the following codes in the sequence shown on Figure 2-1: (1) ESCORE; (2) MICBURN-E; (3) CASMO-2E; (4) NORGE-B; (5) SIMULATE-E; (6) SIMTRAN-E; (7) STRODE; and (8) RETRAN-02. ESCORE is used to provide the fuel rod temperature distribution used in CASMO-2E and the gap conductance used in RETRAN-02. MICBURN-E provides the gadolinia cross sections used in CASMO-2E. CASMO-2E is used to perform the lattice physics analysis to generate the cross sections for SIMULATE-E and the inverse neutron velocity and total effective delayed neutron yield for SIMTRAN-E. NORGE-B is used to transfer the CASMO-2E data to SIMULATE-E and SIMTRAN-E. SIMULATE-E develops the three-dimensional macroscopic cross section data to be processed by SIMTRAN-E. SIMTRAN-E collapses the three-dimensional cross section data to one dimension and transfers the other nuclear parameters to RETRAN-02. STRODE is used adjust the moderator density feedback behavior and delayed neutron fraction data for input to RETRAN-02. RETRAN-02 is used to perform the transient analysis.

5.3.13.4 Licensing Analysis Procedure

In the Code overpressure protection analysis, the following conditions are applied.

(a) The analysis is performed at the most limiting point on the power/flow operating map, consistent with the license basis assumption on maximum power level.

(b) The normal scram times are used.

(c) Scram time delay is the maximum technical specification value.

(d) RPT time delay is the maximum technical specification value.

(e) The six safety/relief valves with the lowest opening setpoints are assumed to be inoperable. Safety/relief valve opening setpoints are consistent with the self-actuated (spring) operation mode.

(f) The analysis is performed at end of cycle conditions

(g) The fastest MSIV closing time identified in the technical specifications is used in the analysis.

Events caused by low reactor water level trip setpoints, including ECCS initiation, are not included in the simulation. These events, should they occur, will be later than the time of challenge or nearest approach to the event acceptance limits has occurred.

5.3.13.5 Sensitivity Studies/Justification of Procedure

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. Even further conservatism is incorporated by the use of upset events limits , rather than the more appropriate emergency or faulted limits for an event of the probability analyzed. No further sensitivity studies are required.

5.3.13.6 Typical Results

From the method described in Section 5.3.13.4, an analysis of the main steam isolation valve closure (MSLVC) transient was performed for Cycle 4 for normal scram speed with RPT. Plots of core power, core average heat flux, core inlet flow, reactor vessel steam dome pressure, vessel steam flow, reactor vessel water level, and feedwater flow are provided on Figures 5.3.13-1 through 5.3.13-7. For this event, the figure of merit for the analysis is peak reactor vessel pressure. The peak calculated pressure is 1313 psig, which is below the limit of 1375 psig.

A comparison of the Supply System and fuel supplier (ANF) Code overpressure protection system analysis results is provided on Table 5.3.13-2. There is good agreement between the two results with respect to the event limit.

Table 5.3.13-1

Sequence of Events for the
Code Overpressure Protection Analysis

Time Secs	Events
0.0	Closure of all MSIVs Begin.
1.81	Scram Initiated on High Neutron Flux.
3.00	All MSIVs are fully closed.
3.09	Group 3 safety/relief valves actuated.
3.7	Peak reactor vessel pressure is reached.
4.10	Group 4 safety/relief valves actuated.
4.15	Group 5 safety/relief valves actuated.
4.17	Recirculation pumps trip on high reactor vessel pressure.

Table 5.3.13-2

Results of Code Overpressure Protection Analysis

	Supply System Results	Fuel Supplier Results
Peak Neutron Flux (% rated)	418	669
Peak Heat Flux (% rated)	132	133
Peak Reactor Vessel Pressure (psig)	1313	1315

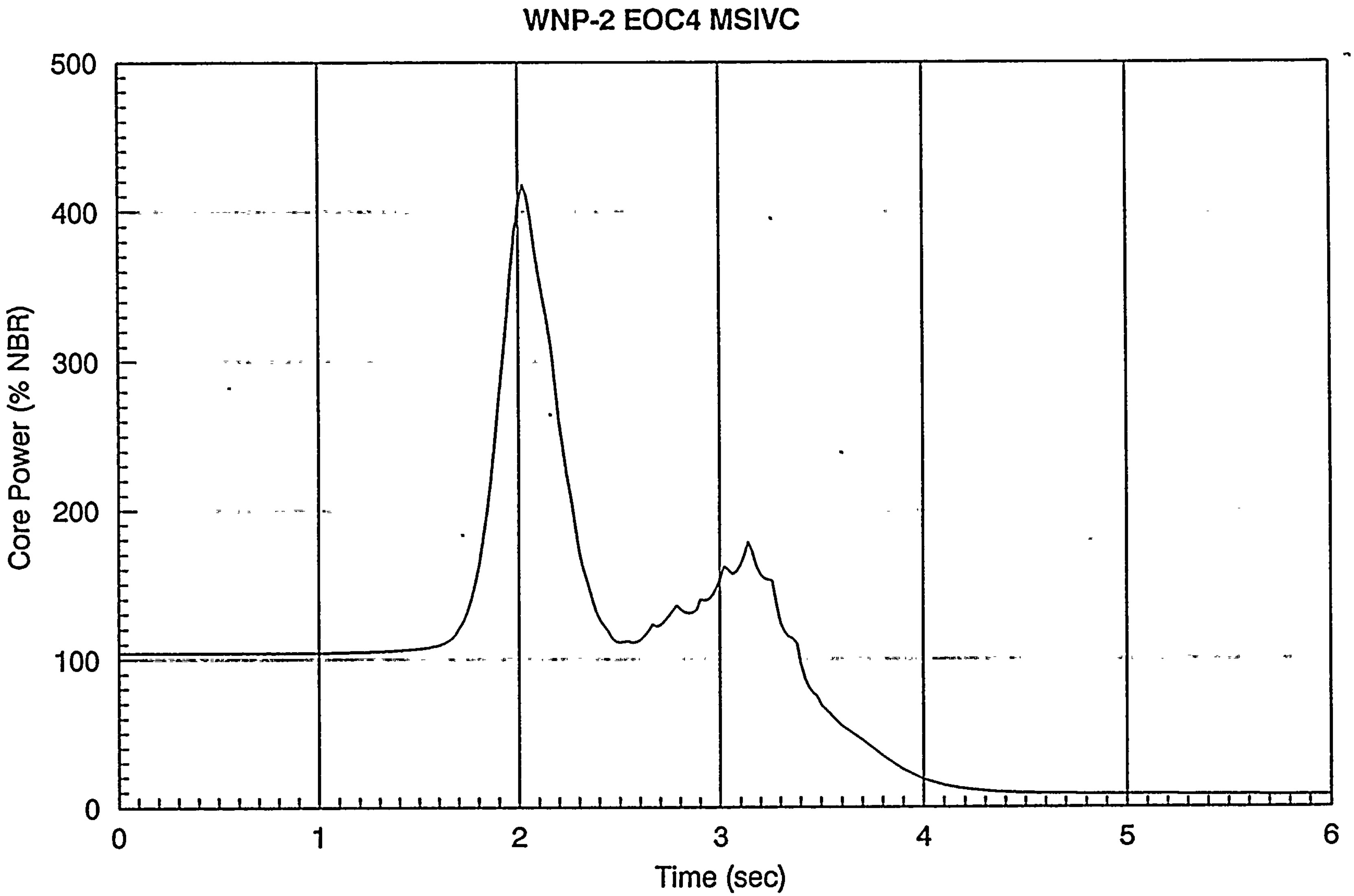

Figure 5.3.13-1 MSIV Closure Results, RPT, Normal Scram Time

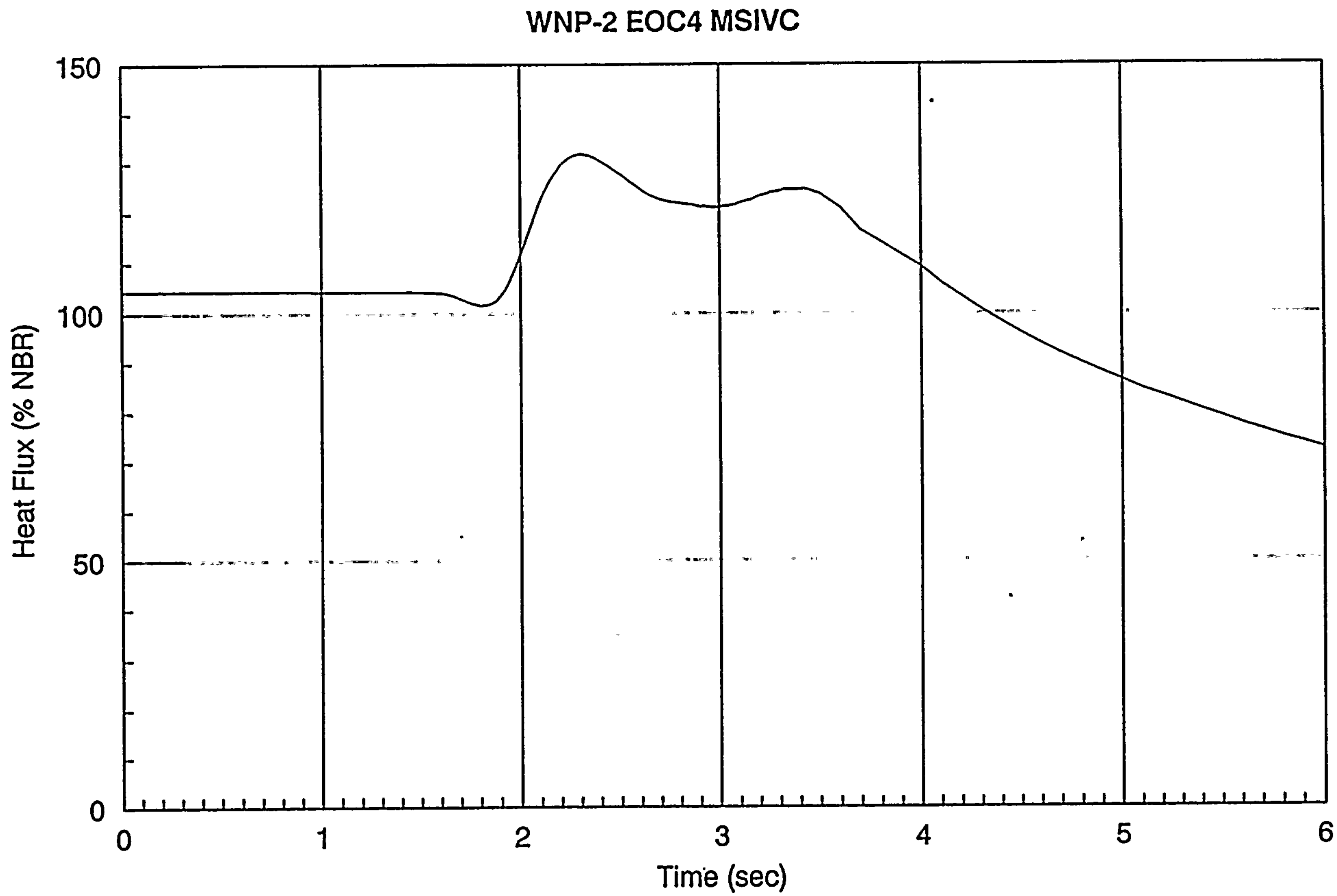

Figure 5.3.13-2 MSIV Closure Results, RPT, Normal Scram Time

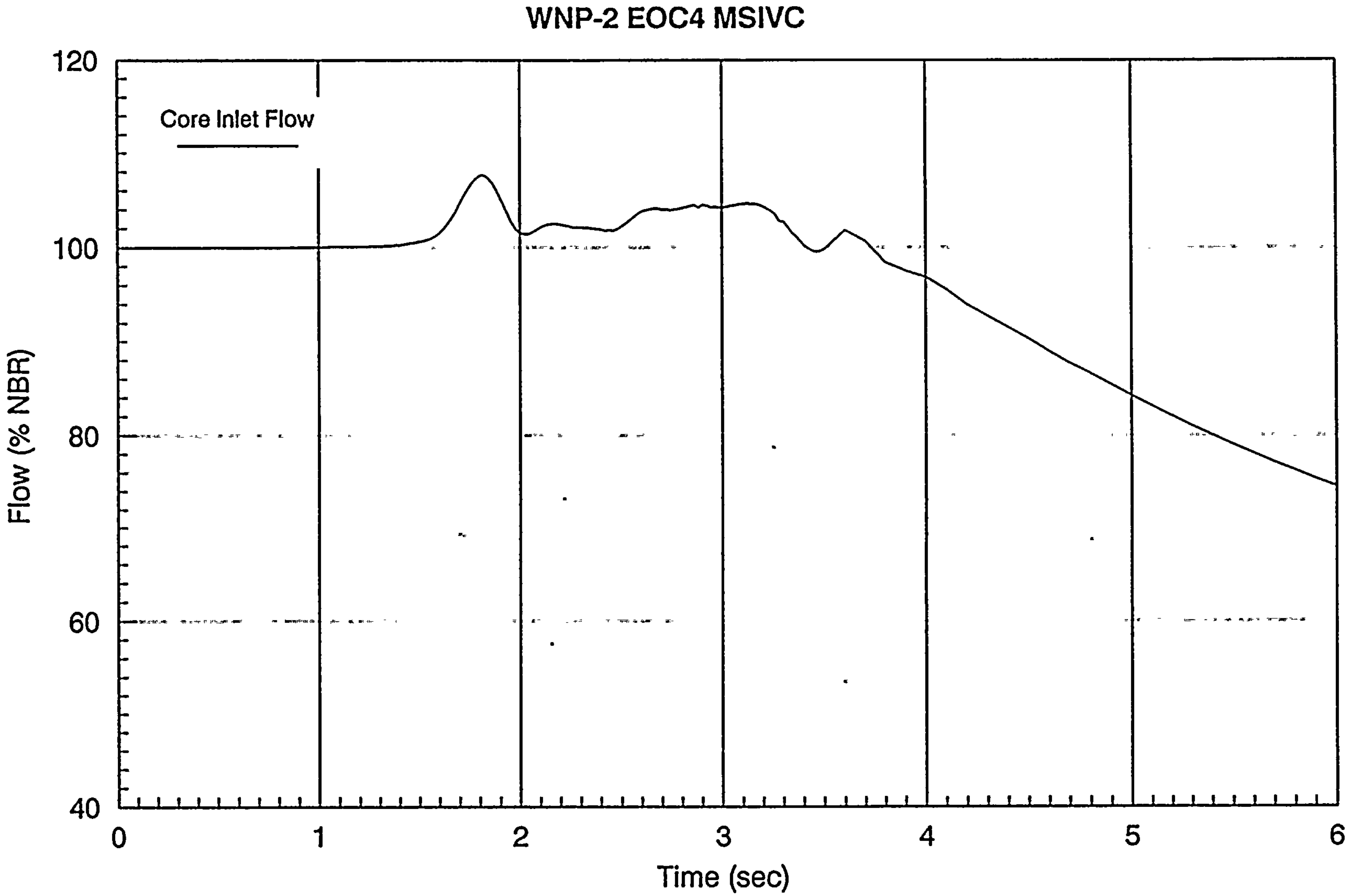

Figure 5.3.13-3 MSIV Closure Results, RPT, Normal Scram Time

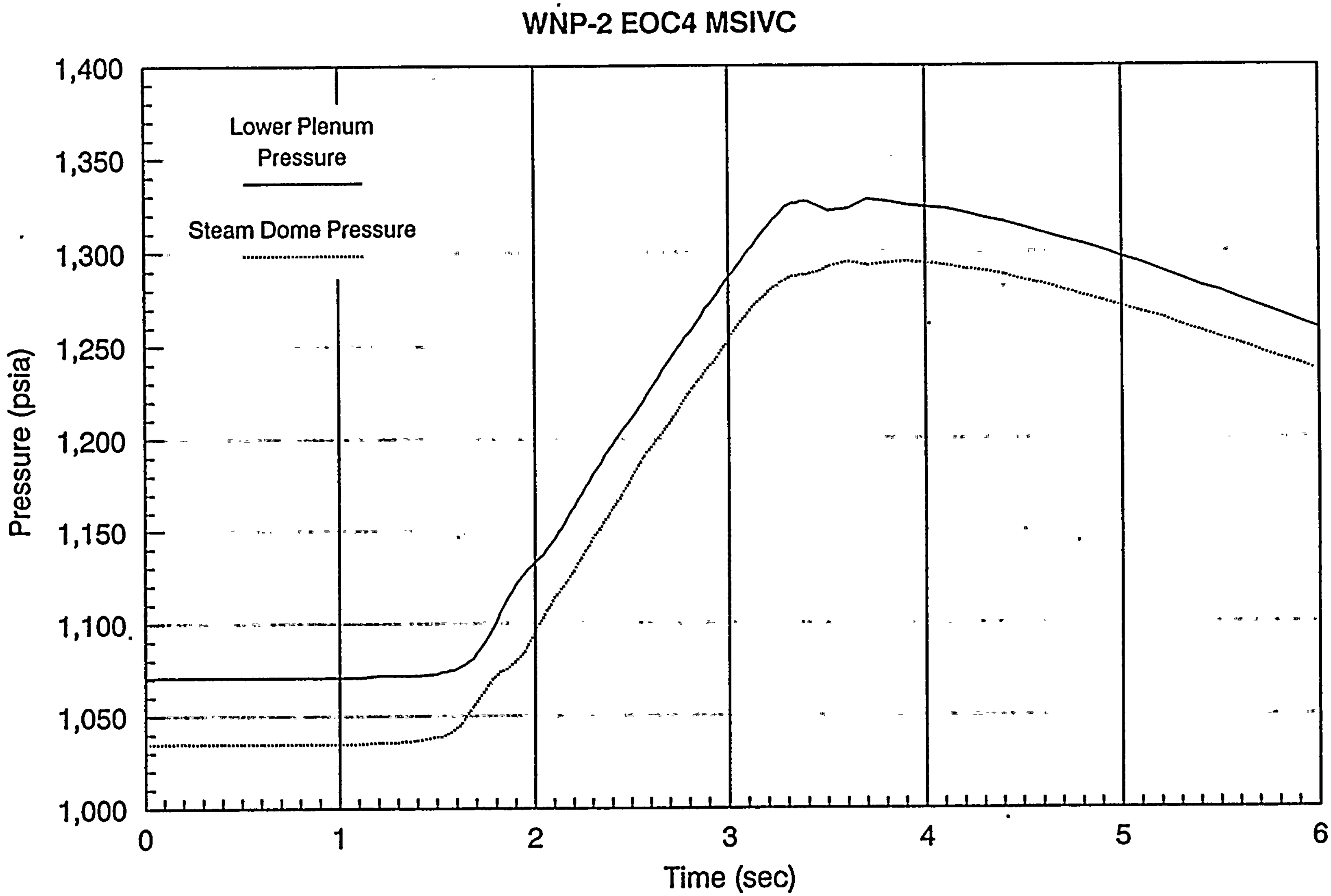

Figure 5.3.13-4 MSIV Closure Results, RPT, Normal Scram Time

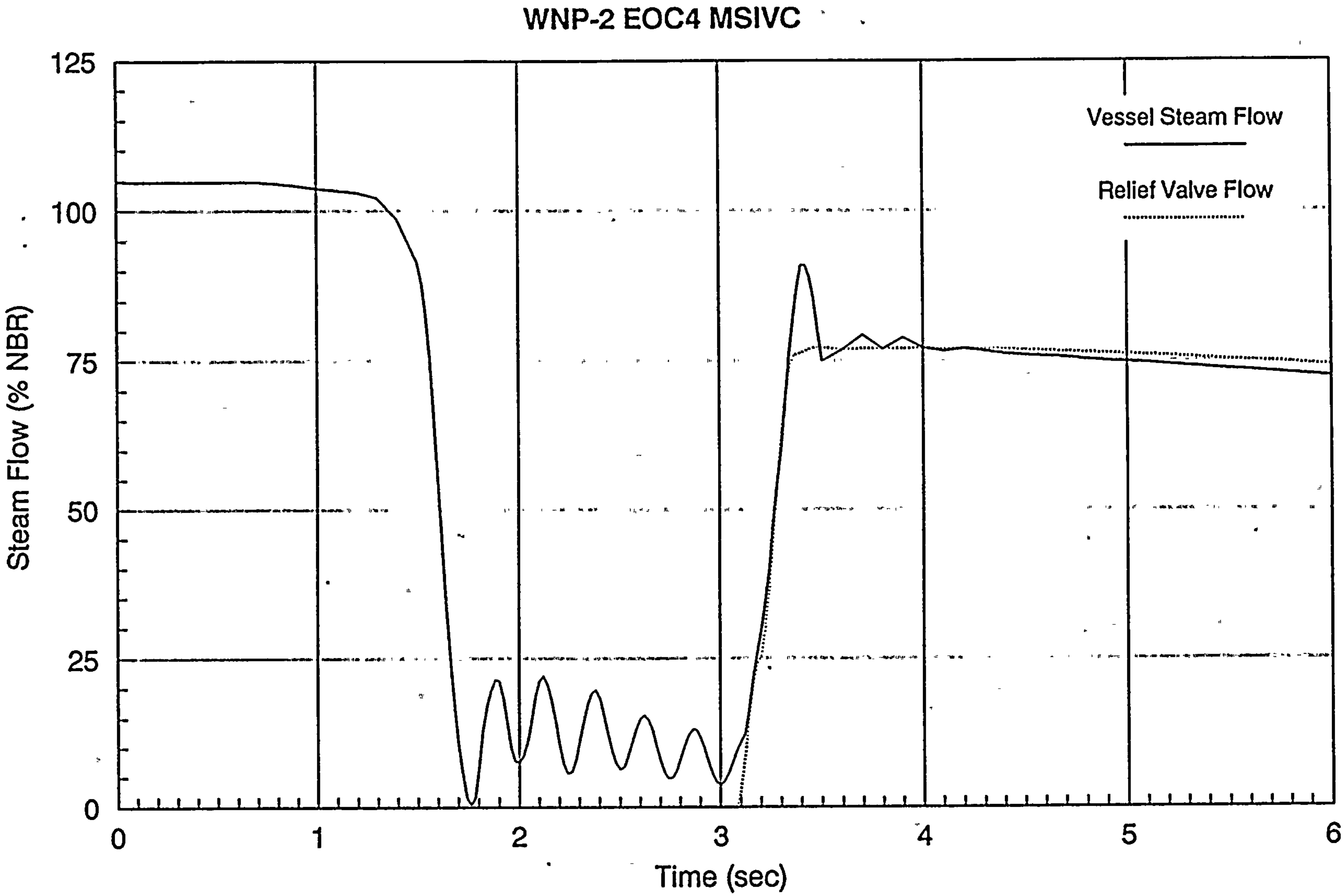

Figure 5.3.13-5 MSIV Closure Results, RPT, Normal Scram Time

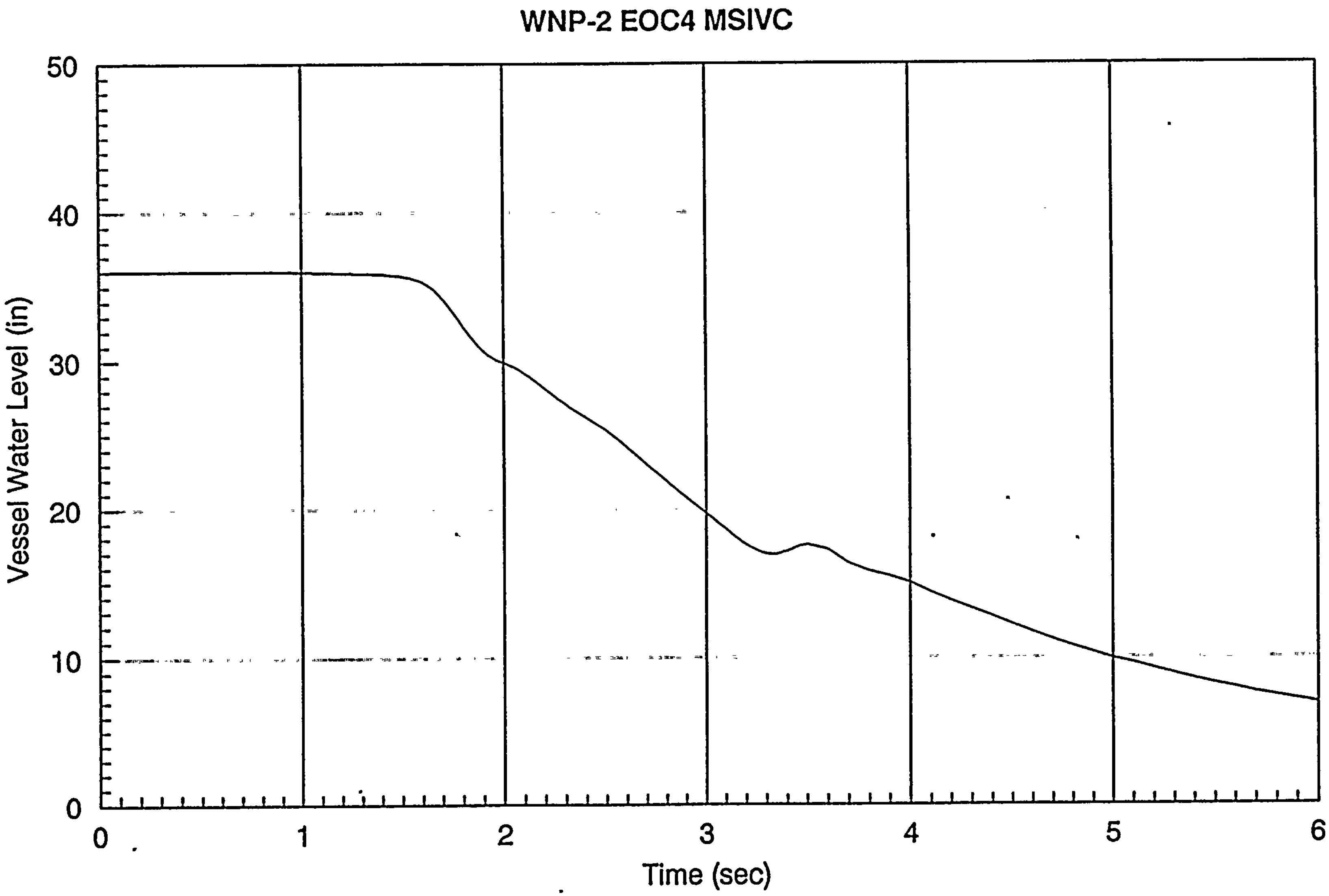

Figure 5.3.13-6 MSIV Closure Results, RPT, Normal Scram Time

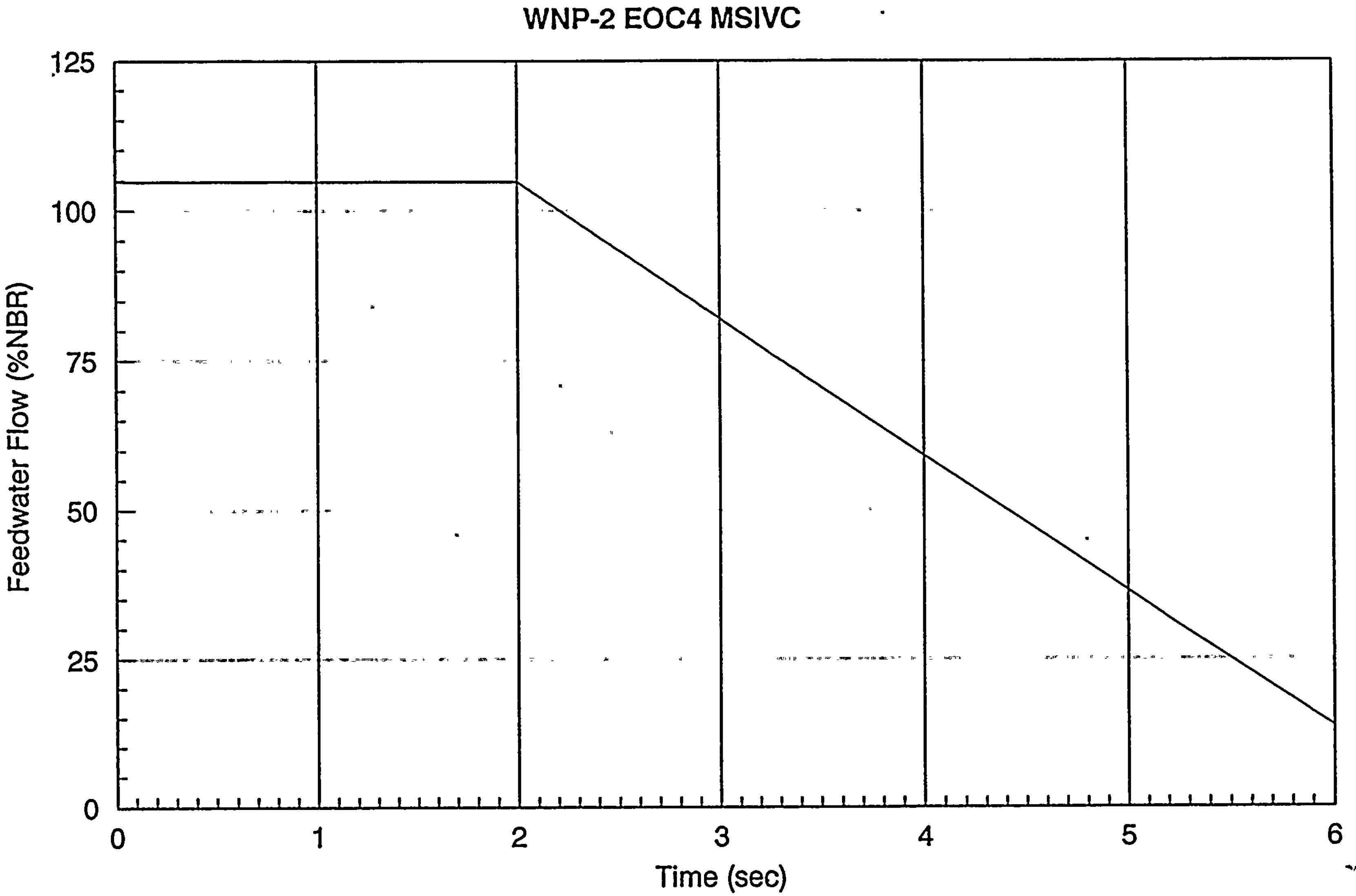

Figure 5.3.13-7 MSIV Closure Results, RPT, Normal Scram Time

5.3.14 Stability

5.3.14.1 Event Description

Stability evaluations are performed to demonstrate that the reactor core and associated coolant, control, and protection systems are designed to assure that power or flow oscillations which can result in conditions exceeding SAFDLs either are not possible or can be detected readily and suppressed. There are three general types of stability considered in the design and analysis of BWR systems:

- o Stability of the entire reactor system in response to changes in system pressure, flow, and water level as determined by the coupled response of the overall plant dynamics and the turbine and reactor control systems (total plant stability).

- o Stability of the reactor core in response to changes in core flow, subcooling or pressure, including nuclear feedback effects from changes in core voids and fuel temperature (core stability).

- o Thermo-hydrodynamic stability of individual fuel channels at various power and flow conditions due to perturbations in flow or channel boundary conditions, independent of reactor system controls or nuclear feedback (channel hydrodynamic stability).

Of these three types of stability, only the core and channel stability evaluations are of importance in the reload fuel analysis process. The total plant stability was demonstrated during the plant startup test program, and the introduction of reload fuel does not significantly change the overall plant dynamics and control system performance.

The potential for encountering instabilities in a BWR varies with the plant operating condition. Acceptable plant performance with respect to plant stability is to be demonstrated over the allowable

operational regions of the power/flow map. Stability margin decrease in the low flow and high power regions of the power/flow map. This is the operating region of the power/flow map that is the subject of stability evaluations in the reload fuel analysis process. Because stability is associated with perturbations (noise) encountered during normal plant operation, no particular initiating event is associated with the stability evaluations.

5.3.14.1.1 Initial Conditions and Operational Assumptions

The following operational conditions and assumptions form the principal bases for core and channel hydrodynamic stability evaluations.

(a) The reactor is operating at a high power level and low core flow rate within the allowable power/flow operating map.

(b) A perturbation in the operating state occurs.

5.3.14.1.2 Operator Actions

There are no operator actions associated with the evaluation of this event because it is a demonstration of the acceptability of the allowable regions in the power/flow map. Should the plant encounter an instability, the operator would be expected to take the actions identified in the plant technical specifications.

5.3.14.1.3 Event Acceptance Limits

The acceptance limit for this evaluation is that there is no predicted instability within the allowable power flow map identified in the plant technical specifications.

5.3.14.2 Evaluation Considerations

This section describes the key evaluation considerations applicable to the stability. It includes: (1) a description of the phenomena occurring during the event that have a significant impact on the event consequences; (2) a discussion of the system performance characteristics that can significantly affect the course of the event; and (3) a discussion of the performance characteristics of the important components as they relate to the event consequences.

5.3.14.2.1 Key Phenomena

Only the core and channel hydrodynamic phenomena are important in the evaluation of the stability phenomena. This section describes these phenomena as they relate to stability evaluations.

<u>Core Phenomena</u> The important core phenomena associated with stability are the fuel time constant, the rate at which voids are removed from the core (void sweep time), and reactor kinetics parameters. The closed loop neutron flux frequency response exhibits a resonance whose frequency is related to the void sweep time/fuel time constant relationship and the void reactivity. The core thermal power exhibits a resonance at the same frequency, but the magnitude is much less due to the attenuation from the fuel time constant. For a given fuel design, the most significant effects on the core stability are due to the change in void sweep time with the state of the core and a more negative void reactivity with fuel exposure. Fuel design changes that increase the gap conductance and heat transfer area can also have a significant effect on core stability. These changes cause a decrease in phase and gain margins, and consequently, tend to destabilize the system.

<u>Channel Hydrodynamic Phenomena</u> The predominant type of hydrodynamic instability that can occur in a BWR channel is one which arises from a dependency of the vapor volume production rate upon the flow rate as a result of momentum, mass, and energy conservation. Void volume depends on mass flow rate because of

energy and mass conservation; however, mass flow rate simultaneously depends on void volume because of momentum conservation. Channel stability is sensitive to the axial power distribution and channel pressure drop characteristics.

5.3.14.2.2 Systems Considerations

The plant system performance characteristics do not effect the core and channel stability evaluation.

5.3.14.2.3 Component Performance Characteristics

The only component performance characteristics of importance in the stability evaluation is the fuel design and its relationship to the core design. Changes in the fuel design as they impact the key phenomena can have a significant effect on the core stability evaluation.

5.3.14.3 Methodology/Integration of Codes and Analysis

The plant technical specifications contain requirements that restrict plant operation within the power/flow map. The restrictions are consistent with the General Electric Company recommendations contained in Service Information Letter (SIL) 380. In addition, a stability monitoring system has been incorporated into the WNP-2 plant design. Technical specification action statement are provided to assure compliance with the event acceptance limits associated with stability. No further evaluations relative to stability are required as long as the technical specification bases are applicable.

5.3.14.4 Licensing Evaluation Procedure

For reload fuel evaluation purposes, it is necessary to determine that the plant technical specification bases related to stability are applicable.

5.3.14.5 Sensitivity Studies/Justification of Procedure

Sensitivity studies are not required to demonstrate compliance with the technical specifications.

5.3.14.6 Typical Results

No specific analyses are required to demonstrate compliance with the technical specifications.

SECTION 6.0
DOCUMENTATION REQUIREMENTS

The Supply System must appropriately document operating limits and other technical specifications that may change during the lifetime of the reactor. This section describes that documentation.

6.1 RELOAD SUMMARY REPORT

The reload summary report documents the results of the analyses of the potentially limiting events. These events are summarized in table 5.2-1. Table 6-1 provides an outline for reload summary reports. This outline may be updated as required to reflect changes in NRC requirements.

6.2 CORE OPERATING LIMITS REPORT

The NRC has granted the Supply System approval to use a core operating limits report (COLR) to specify most cycle-specific fuel and core operating limits for plant operation. (See NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications.") The NRC has established three requirements:

- the addition of a named report, the COLR, that lists the values of the cycle-specific parameter limits that have been established using an NRC-approved methodology; these values must be consistent with all applicable safety analysis limits
- the addition of an administrative reporting requirement to submit the COLR to the NRC
- the modification of individual technical specifications to note that cycle-specific parameters are to be maintained within the limits specified in the COLR, and that the COLR must identify approved reports that describe the development of the cycle-specific limits.

The Supply System COLR specifies operating limits for the APLHGR, MCPR, and LHGR.

6.3 TECHNICAL SPECIFICATIONS

The safety limit minimum critical power ratio (SLMCPR) and many instrument setpoints remain in the technical specifications. The SLMCPR may change from one operating cycle to the next. Setpoints in the technical specifications could also change. The reload summary report will identify any changes made to the SLMCPR or to instrument setpoints; it will also document the analyses that provide the bases for these changes. The Supply System will then seek any necessary amendments to technical specifications.

Table 6-1
Outline of the Reload Summary Report

1.0 Introduction

Describes the cycle this reload analysis is for, the types of fuel assembly designs in this core loading, and any distinguishing characteristics of the cycle.

2.0 Fuel Mechanical Design Analysis

References applicable fuel vendor design reports for fuels resident in the core in this cycle. Compares expected power histories for the fuel designs with power histories in the vendor fuel mechanical design analyses.

3.0 Thermal-Hydraulic Analysis

3.1 Fuel Centerline Temperature

Lists all fuel vendor references for the LHGR associated with the fuel centerline melting point. Provides evidence of the margin to centerline melting.

3.2 Bypass Flow

Provides the bypass flow for the maximum rod line at maximum flow.

3.3 MCPR Fuel Cladding Integrity Safety Limit

Lists the core power, core inlet enthalpy, steam dome pressure, feedwater temperature, and design basis radial and local power distributions.

Table 6-1 (Continued)
Outline of the Reload Summary Report

4.0 Nuclear Design Analysis

4.1 Fuel Assembly Nuclear Design

Lists the average enrichment, radial and axial enrichment distributions, burnable absorber rods, non-fueled rods, and neutronic design parameters for each new assembly design in the cycle.

4.2 Core Nuclear Design

Describes the core loading pattern. Lists the core exposure at the end of the previous cycle, the core exposure at the beginning of this cycle, and the core exposure at the end of this cycle. Also lists the following core reactivity characteristics:

- o beginning-of-cycle cold target $k_{effective}$
- o beginning-of-cycle cold $k_{effective}$ with all control rods out
- o beginning-of-cycle cold $k_{effective}$ with the strongest control rod out
- o reactivity defect (R-value) in terms of Δk
- o Δk for the standby liquid control system at 660 ppm boron
- o decay ratio (included only if analysis evaluates core hydrodynamic stability).

5.0 Anticipated Operational Occurrence Analyses

5.1 Anticipated Operational Occurrences at Increased Core Flow Conditions

Lists the limiting transients at increased core flow conditions and indicates where on the power flow map the transients are initiated. Evaluates transients at increased flow conditions relative to design conditions. Discusses assumptions made for specific transients (i.e. RPT in service or out of service, scram speed, exposure conditions, and final feedwater temperature reduction). Tabulates events, power and flow conditions, maximum heat flux, maximum power, maximum pressure, and ΔCPR for the various fuel types.

Table 6-1 (Continued)
Outline of the Reload Summary Report

5.2 Reduced Flow Operation

Lists the limiting transients at reduced flow conditions and indicates where on the power flow map the transients are initiated unless transients in 5.1 are bounding. Tabulates the results for these transients as in 5.1. Provides the reduced flow MCPR operating limit.

5.3 Reduced Power and Single Loop Operation

Lists and tabulates any transients that are more severe at reduced power than at full power. Provides a reduced power MCPR operating limit. Lists and tabulates transients for single loop operation and gives the corresponding MCPR operating limits and MAPLHGR limits.

5.4 ASME Overpressurization Evaluation

Discusses the limiting event, the worst single failure, maximum vessel pressure, and steam dome pressure.

5.5 Control Rod Withdrawal Error

Illustrates the initial control rod pattern for the control rod withdrawal error analysis. Tabulates the RBM setting, the distance the rod is withdrawn, and the corresponding limiting ΔCPR.

5.6 Determination of Thermal Margins

Presents a summary of MCPR operating limits for each fuel type over various exposure ranges of the cycle.

Table 6-1 (Continued)
Outline of the Reload Summary Report

6.0 Postulated Accidents

6.1 Loss-of-Coolant Accident

Presents the results of vendor analyses of peak clad temperature and peak local metal-water reaction for each fuel type given the Supply System local peaking factors, fuel rod power histories, void coefficient, Doppler coefficient, and time-dependent scram reactivity.

6.2 Control Rod Drop Accident

Presents the results of vendor analyses of maximum deposited fuel rod enthalpy given Supply System results for maximum control rod worths, four-assembly local peaking factors, Doppler coefficient, and delayed neutron fraction.

6.3 Fuel Loading Error

Lists the ΔCPR for the worst errors in locating and/or rotating fuel assemblies.

6.4 Fuel Handling Accident

Lists the radiological consequences of a fuel handling accident (FHA) in terms of whole body dose and thyroid dose for every fuel design in the core. This data will come from fuel design analyses performed by the fuel vendor.

Table 6-1 (Continued)
Outline of the Reload Summary Report

7.0 Technical Specifications

7.1 Limits

Presents the SLMCPR and the steam dome pressure safety limit.

7.2 Limiting Conditions for Operation

Lists the following limiting conditions for operation:
- APLHGR limits versus exposure for each fuel type
- MCPR limits for each fuel type for given exposure ranges and off-normal conditions
- LHGR limits versus exposure for each fuel type
- Surveillance requirements, if required, for scram insertion time and stability.

8.0 References

Lists all items referenced in the summary report.

SECTION 7.0
LOADING PATTERNS

The reload analysis must be completed long enough before fuel reloading begins to allow time for manufacturing fuel and meeting safety analysis requirements. The analysis schedule requires making some assumptions about what the core characteristics will be at the end of the current cycle so that work can begin on a reference loading pattern for the new cycle. The core characteristics at the end of the cycle may differ, however, from the characteristics postulated during the reload analysis. If the difference is large enough, some parameters in the safety analysis will need to be rechecked. If the differences require deviation from the reference loading pattern, additional parameters will also need to be checked.

7.1 CHANGES IN REFERENCE LOADING PATTERN

The following parameters must be checked if the core characteristics at the end of cycle differ significantly from the postulated characteristics and/or if the actual loading pattern deviates from the reference loading pattern.

7.1.1 End-of-Cycle Exposure and Axial Exposure Distribution

The reference loading pattern assumes a certain core average exposure and axial exposure distribution at the end of the current operating cycle. Either of the following requires evaluation to determine the impact on the event analyses:

- o any deviation beyond specified maximum and minimum core average exposure values assumed for the reference loading pattern
- o any significant deviation from the assumed axial exposure distributions used in these evaluations.

7.1.2 Number and Location of Reload Fuel Assemblies

The reference loading pattern specifies the number and location of fresh fuel assemblies in each reload. Any increase in number or change in location requires re-examinination of the reload analyses. If the number of fresh assemblies is decreased, the reductions may be made only in peripheral, control-rod-centered, four-fuel-assembly cells. Furthermore, the number of fresh fuel assemblies eliminated may not exceed 10% of the reload batch or 2% of the total core, whichever is smaller. Failing to meet this criterion requires a re-examination of the reload analyses.

7.1.3 Type, Number, and Location of Exposed Fuel Assemblies

The reference loading pattern specifies the type, number, and location of exposed assemblies in the reload. Fuel assemblies of a different type may be substituted as long as they are less reactive. Any changes in the placement of fuel assemblies should be in the regions of least importance, and replacements should match as closely as possible the exposure and exposure history of the assemblies they are replacing.

7.2 RE-EXAMINATION OF A RELOAD ANALYSIS

Re-examination of a reload analysis requires re-evaluating the following:

- o scram reactivity
- o void reactivity
- o shutdown margin
- o generic parameters for a rod drop accident
- o misloaded fuel assembly ΔCPR
- o RBM response for a control rod withdrawal error
- o limiting parameters for core-wide transients

These parameters are of primary importance in establishing a core operating limit or in determining the consequences of an analyzed

event. Other parameters affected by the core loading pattern are less significant and are covered by conservatism in the analysis process.

SECTION 8.0
CONCLUSIONS

This applications topical report describes the Supply System reload analysis methodology. This methodology, when integrated with the reload fuel supplier methodology, can perform all tasks and evaluations required for the reload analysis. The reload summary report submitted at the end of each reload analysis will ensure that all requirements have been met.

This report specifically documents the following:

1. The Supply System reload analysis methodology uses conservative procedures that ensure successful reload fuel designs and core configurations.

2. The spectrum of events considered in the reload analysis will be consistent with current regulatory requirements and with current industry practice.

3. The event limits used in the reload analysis will be conservative.

4. Each of the codes used in the reload analysis can perform the required function; the integrated set of codes can perform the event analyses and provide the data required by the fuel supplier.

5. The Supply System methodology will ensure thermal-hydraulic and nuclear compatibility between the resident fuel and reload fuel assemblies.

6. The Supply System input to the fuel vendor methodology will establish a conservative MCPR that ensures fuel cladding integrity.

7. The Supply System evaluation of PAFF limits provided by the fuel supplier will ensure fuel integrity.

8. The reload analyses will identify any necessary changes in core operating limits and technical specifications.

9. Startup physics testing will confirm the suitability of the fuel loading pattern.

10. The reload summary reports will satisfy all documentation requirements specified in 10CFR50.59.

This report documents that the Supply System reload analysis methodology will provide conservative reload fuel designs and core configurations that ensure the safety of the facility and the surrounding community.

APPENDIX A

STATISTICAL COMBINATION OF UNCERTAINTIES METHODOLOGY APPLICATION

In the plant safety analysis process, it is necessary to account for uncertainties in the model and model inputs, operating state, and instrumentation systems. These uncertainties in the safety analysis process are typically treated through a conservative or deterministic approach. The deterministic approach to the treatment of uncertainties involves taking the principal uncertainty components at an arbitrarily defined adverse bound, and no attempt is made to estimate the overall uncertainty distribution. This approach essentially assumes that all significant uncertainties are simultaneously at their most adverse values within the possible range of conditions, which results in very conservative and unlikely predictions of event consequences.

In most cases, the plant has sufficient margin to accommodate this approach and no further analyses are required. However, the use of the deterministic approach can lead to core operating limits or technical specification setpoints that result in undesirable plant restrictions. For those cases, the WNP-2 reactor analysis methodology incorporates the use of a statistical combination of uncertainties (SCU) methodology to define a set of more operationally acceptable setpoints while retaining an appropriate level of conservatism.

In the SCU approach, direct stochastic analysis of the event, using the event analysis methodology, requires an unacceptable expenditure of resources. Therefore, response surface techniques are used in the event simulation to reduce the resource requirements to an acceptable level.

A response surface is an algorithm that approximates the event analysis codes. There are five basic steps associated with the response surface methodology: (1) the determination of the model uncertainty; (2) the quantification and selection of the parameters

to be used in the response surface; (3) the development of the response surface; (4) the simulation of the event using the response surface; and (5) the convolution of the response surface for the event with the model uncertainty. Each of these steps is described below.

A.1 DETERMINATION OF MODEL UNCERTAINTY

To determine the model uncertainty, studies are performed to establish the effect of key modeling and model input parameter uncertainties on the event analysis results. In these studies, the analysis model is run with perturbed inputs to quantify the effect of changes in the model and model inputs. The results of the model uncertainty studies are statistically combined to determine the overall model uncertainty. The model uncertainty established through this process is then combined with the results of the response surface analysis to provide the overall event analysis result probabilistic statement.

The Supply System model uncertainty procedure has been applied to the generator load rejection without bypass event (See Section 5.3.3) which is the limiting transient with respect to minimum critical power ratio (MCPR). The base case in the model uncertainty evaluation uses technical specification scram time and recirculation pump trip (RPT) at licensing basis power level and 100% core flow. The parameters considered in the model uncertainty evaluation analysis fall into five groups: (a) Nuclear Model Parameters; (b) Core Thermal-Hydraulics Parameters; (c) Recirculation System Parameters; (d) Steam Line Model Parameters; and (e) Vessel and Loop Geometry Parameters.

The effect of the uncertainties in each of these parameters on the calculated consequences of the generator load rejection without bypass event is shown in Table A-1. RCPR (defined as ΔCPR/initial CPR) is used as the figure of merit for the uncertainty evaluation. The change in RCPR in each uncertainty evaluation is used to quantify the model uncertainty affect on the transient thermal

margin. The overall model uncertainty is determined by taking the square root of the sum of the squares of the ΔRCPR due to the individual model and input parameter uncertainties.

A.1.1 Nuclear Model Parameter Uncertainties

Nuclear model uncertainties affect the coupling of the core power generation to the moderator conditions and include: moderator void reactivity feedback; fuel temperature feedback (Doppler); scram reactivity; and fraction of prompt energy deposited directly in the coolant. The magnitude of the parameter uncertainties and their effects on the calculated ΔRCPR is shown in Table A-1.

Increasing the moderator void reactivity feedback increases the severity of the event because the power generation will increase, relative to the base case, for an equivalent core void fraction reduction. The sensitivity evaluation case used a void reactivity feedback 13% more negative than the base case. This conservatively bounds the uncertainty in void reactivity (Reference A1). As seen from Table A-1, this is the largest single source of overall model uncertainty.

Decreasing the Doppler fuel temperature reactivity feedback increases the severity of the event because it reduces the mitigation of power generation due to increased fuel temperature. The sensitivity evaluation case reduced the Doppler reactivity feedback by 10%, consistent with the NRC staff recommendation (Reference A2, Page II-32). The contribution to overall model uncertainty is small.

The prompt moderator heating fraction is the portion of the total fission energy that is deposited directly into the core coolant by gamma heating. Reducing the prompt moderator heating fraction increases the severity of the event because it increases the rod heat flux for the same fission power. From Reference A2, for typical BWR fuel the 95% confidence lower limit is about 20% lower than the nominal value. For additional conservatism, a 25%

reduction was used in the Supply System analysis. As seen from Table A-1, this is a significant contribution to the overall model uncertainty.

Decreasing the control rod scram reactivity feedback increases the severity of the event because it reduces the mitigation of power generation due to control rod insertion. The sensitivity evaluation case reduces the scram reactivity feedback by 10%, consistent with the NRC staff recommendation (Reference A2, Page II-32). The contribution to overall model uncertainty is small.

A.1.2 Core Thermal-Hydraulic Parameter Uncertainties

Core thermal-hydraulic model uncertainties include: drift flux model correlation parameters; core pressure drop loss coefficients; bypass flow fraction; fuel pin conduction heat transfer; and core power. These parameters directly affect the core moderator conditions during the transient. The magnitude of the parameter uncertainties and their effects on the calculated ΔRCPR is shown in Table A-1.

The drift flux model parameters of interest are the concentration parameter, C_o, and the drift velocity, V_{gj}. For uncertainty evaluation, the NRC staff recommends using a limiting value of 1.0 for C_o, and a 30% variation in V_{gj}. These cases were performed in the RETRAN-02 sensitivity studies by variation of the parameters KAPPA1 and CGL, respectively. The contribution of drift flux parameter uncertainty to overall model uncertainty is small.

The Dittus-Boetler heat transfer correlation is used in the profile fit model to calculate the subcooled voids for the neutronic feedback. To study the impact of uncertainty on the calculated ΔRCPR, the leading coefficient in the correlation, CDB, is increased by 20% as recommended in Reference A3. The contribution of correlation uncertainty to overall model uncertainty is small.

The Hancox-Nicoll heat transfer correlation is also used in the profile fit model to calculate the subcooled voids for the neutronic feedback. To study the impact of uncertainty on the calculated ΔRCPR, the leading coefficient in the correlation, CHN, is increased by 20% as recommended in Reference A3. The contribution of correlation uncertainty to overall model uncertainty is small.

The core dynamic pressure drop loss coefficients were reduced by 20% as recommended by the NRC staff (Reference A2, Page II-32). The contribution to overall model uncertainty is small.

From Reference A4, the uncertainty in initial core bypass flow fraction is less than 20%. The initial bypass flow fraction in this study was reduced 20% from the base case. The contribution to overall model uncertainty is small.

Uncertainties in the overall fuel rod thermal time constant contribute to the overall model uncertainty. to investigate the sensitivity, the fuel pellet and cladding radial noding was increased by a factor of two. The contribution to overall model uncertainty is small.

General Electric has made extensive studies of the core power uncertainties (Reference A2) based on analysis of the uncertainties of components in the plant energy balance. They concluded that the one standard deviation uncertainty is 2%. The Supply System has used an uncertainty of 4% in the model uncertainty evaluation. The contribution to overall model uncertainty is small.

A.1.3 Recirculation System Parameter Uncertainties

Recirculation system model uncertainties affect the timing and attenuation of the pressure wave from the vessel steam dome to the reactor core and the rate of change of core flow following recirculation pump trip (RPT). The parameters of interest include: recirculation loop inertia; recirculation loop head; jet pump

inertia; steam separator inertia; and the jet pump loss coefficients. The magnitude of the parameter uncertainties and their effects on the calculated ΔRCPR is shown in Table A-1.

The recirculation pump dynamic characteristics affect the core flow following RPT. Based on the analyses performed by General Electric and accepted by the NRC staff (Reference A5) the recirculation loop inertia is increased 100% in the model uncertainty evaluation. As seen in Table A-1, recirculation loop inertia uncertainty has a moderate contribution to overall model uncertainty.

The recirculation pump head is reduced by 10% in the model uncertainty evaluation. The pump head contribution to overall model uncertainty is small.

Jet pump fluid inertia affects the core flow following RPT as well as the timing and attenuation of the pressure wave from the steam lines to the reactor core via the lower plenum region. From Reference A4, there is a significant uncertainty in jet pump fluid inertia. In the model uncertainty evaluation, the inertia was increased by 100%. The contribution to overall model uncertainty is moderate.

Steam separator inertia affects the timing of the pressure wave from the steam lines to the core via the upper plenum region. sensitivity evaluations were performed for the separator liquid outlet (carryunder) junction inertia and for the separator inlet junction inertia. In accordance with Reference A4, the liquid outlet inertia was increased by 100%. In accordance with Reference A2 (Page Q15-2), the inlet inertia was reduced by 30%. The contribution to overall model uncertainty is small.

The jet pump pressure drop loss coefficients were reduced by 20% as recommended by the NRC staff (Reference A2, Page II-32). The contribution to overall model uncertainty is small.

A.1.4 Steam Line Model parameter Uncertainties

Steam line model uncertainties affect the timing and attenuation of the pressure wave from the turbine control valve to the vessel steam dome and include steam line inertia and steam line pressure drop. Increasing the magnitude of the core pressurization causes greater void collapse and greater positive reactivity insertion due to void reactivity feedback. Reducing the steam line pressure wave transit time reduces the consequences of the event. The magnitude of the parameter uncertainties and their effects on the calculated ΔRCPR is shown in Table A-1.

Based on a study performed by TVA and documented in Reference A4, an uncertainty of 7% was used for the main steam line fluid inertia. Relative to the base case, the timing of the pressure wave reaching the core is delayed but the magnitude of the pressure wave is greater. The net effect is a moderate increase in ΔRCPR.

The pressure loss coefficients in the main steam lines directly affect the attenuation of the pressure wave. The main steam line pressure drop loss coefficients were reduced by 20% as recommended by the NRC staff (Reference A2, Page II-32). The contribution to overall model uncertainty is moderate.

A.1.5 Vessel and Loop Geometry Parameter Uncertainties

Vessel and loop geometry uncertainties affect the timing and attenuation of the pressure wave to the reactor core and include the reactor vessel steam dome and the steam line volumes. The magnitude of the parameter uncertainties and their effects on the calculated ΔRCPR is shown in Table A-1.

Decreasing the reactor vessel steam dome volume will increase the severity of a pressurization event because the pressure wave will reach the reactor core sooner. Based on a study performed by TVA and documented in Reference A4, an uncertainty of 5% was used for

the steam dome volume. Steam dome volume uncertainty is a moderate contribution to overall model uncertainty.

Decreasing the main steam line volume will also affect the severity of a pressurization event because the pressure wave will reach the reactor core sooner. Based on Reference A4, an uncertainty of 5% was used for the main steam line volume. Main steam line volume uncertainty is a small contribution to overall model uncertainty. Reducing the steam line volume slightly decreases the calculated ΔRCPR because, although the pressure wave reaches the core sooner, it has a reduced magnitude because of its more rapid transit time through the steam lines.

A.2 QUANTIFICATION AND SELECTION OF RESPONSE SURFACE PARAMETERS

A screening process aids in the selection of the parameters to be used in the development of the response surface. In this screening process, the uncertainties in the key analysis input parameters are characterized and sensitivity studies are performed to determine their relative impact on the analysis results. Generally, the parameter(s) that can provide the greatest relative improvement in figure of merit for the event being analyzed are included in the response surface.

For the generator load rejection without bypass event, the parameter with the greatest potential improvement in calculated thermal margin is control rod scram speed. Control rod insertion timing is affected by timing of the reactor protection system trip signal (e.g., turbine control valve closure), reactor protection system logic response time, the solenoid de-energizing time, and the control rod drive scram insertion time.

Current safety analysis is performed using scram insertion times contained in the plant technical specifications. The technical specification scram times are quite conservative compared to scram times based on plant data. The actual plant control rod scram times that have been measured since October 1985 are retained in a

data base. The data base includes all unplanned full core scrams as well as the scrams that are performed to satisfy plant surveillance test requirements. The plant scram time data base is sufficient to justify the use of more realistic scram times in WNP-2 plant safety analysis.

A computer program was written to perform a statistical evaluation of the scram times contained in the data base. The program reads the measured times for insertion from fully withdrawn to each of four control rod notch positions (0.75 ft, 2.25 ft, 5.75 ft, and 10.75 ft). The scram insertion time mean and standard deviation is then calculated for each notch position. A range of dates is specified for each calculation so that statistical information may be obtained for a subset of the entire data base.

The data for full scrams is considered to be more relevant to system transient analysis than single rod data. All of the full scram data, from the earliest times in the data base to the present, were used to develop a realistic scram time table. This included 4858 scram time measurements for each of the four notch positions. Table A-2 shows the scram times used in the Supply System analysis of the generator load rejection without bypass.

A.3 RESPONSE SURFACE DEVELOPMENT

The statistical simulation of an event requires a large number of cases to be run to obtain the desired accuracy. Response surface methodology provides an efficient way of reducing the number of system and core thermal hydraulic analyses required to make an accurate statistical statement about the event analysis results. A response surface is developed based on experimental design considerations. The experimental design defines the specific system and core thermal hydraulic cases that must be run to develop the response surface algorithm. The response surface is a polynomial fit to the experimental design cases, that approximates the system and core thermal hydraulic analysis methodology for a specific analysis output parameter over the limited range of

interest. To define the response surface, a least squares fitting technique is used to determine the polynomial fitting coefficients. As a part of this process, the fitting error, which represents the uncertainty introduced in the analysis process by the use of the response surface, is determined.

The Supply System response surface is of the general form:

$$Y = B_o + \sum B_i \eta_i + \sum B_{ii} (\eta_i^2 - C) + \sum_{i \neq j} B_{ij} \eta_i \eta_j \qquad \text{(A-1)}$$

Where:

B_o, B_i, B_{ii}, and B_{ij} are the response surface coefficients;
η_i are the independent variables
Y is the response surface dependent variable

For the generator load rejection without bypass event, only the control rod scram time is used as an independent variable and the response surface assumes the simple form:

$$RCPR = B_o + B_1 T + B_{11} T^2 \qquad \text{(A-2)}$$

As shown on Table A-3, five different scram times were analyzed for the construction of the generator load rejection without bypass response surface. The results of the RCPR evaluations for these case were input to the STARS code (See Section 2.3.3). STARS then calculated the fitting coefficients B_o, B_1, and B_{11}. The coefficients for the RCPR in Table A-3 are:

$B_o = 0.1606$
$B_1 = 0.0143$
$B_{11} = -0.00150$

The response surface provides an excellent fit to the calculated RCPR's with an RMS fitting error of 0.006.

A.4 EVENT SIMULATION USING THE RESPONSE SURFACE

Once the response surface has been defined, the event can be simulated using Monte Carlo techniques. In this process, STARS randomly samples the independent parameters and the response surface fitting error, based on their statistical characteristics, and then calculates the dependent variable (RCPR). Each set of random samples forms one "history" for the statistical evaluation of the response surface. 100,000 histories were run to accurately quantify the probability of a given analysis result for the response surface, considering the fitting error. The calculated RCPR at a 95% confidence level is 0.180. The model uncertainty has not been considered in this evaluation.

A.5 CONVOLUTION OF THE RESPONSE SURFACE AND THE MODEL UNCERTAINTY

The final task is to convolve the response surface analytically with the model uncertainty to establish the overall probability of the event analysis results. This probability distribution is then used to demonstrate compliance with the event acceptance limits. This analysis considers the response surface fitting error as well as the model uncertainty discussed in Section A-1. The calculated RCPR at a 95% confidence level is 0.192.

In the STARS analysis, the response surface development, the event simulation using the response surface, and the convolution of the response surface with the overall model uncertainty is performed in a single computer run. The steps have been described separately to clarify the statistical analysis process.

A.6 APPLICATION OF THE SCU METHODOLOGY TO OPERATING LIMIT MCPR

The WNP-2 plant measured scram speed must be consistent with the scram speed used in the statistical approach. The WNP-2 technical specifications contain surveillance requirements that all control rods must be scram tested after each refueling outage and that 10% of the control rods must be test at 120-day intervals. The

surveillance testing data is utilized to compute τ_{av}, the average scram time to control rod notch position 39 (the position closest to 20% insertion, which is the portion of the scram that most affects a limiting pressurization transient). The average scram time to notch 39 τ_{av}, is updated after each surveillance testing.

The average scram time, τ_{av}, is then tested against the analysis mean using

$$\tau_{av} \le \tau_B \tag{A-3}$$

where

$$\tau_B \equiv \mu + A\,\sigma \tag{A-4}$$

$$A \equiv 1.65\ (N/n)^{1/2}$$

The parameters μ and σ are the mean and standard deviation of the distribution for the scram insertion time to control rod notch position 39 used in the WNP-2 SCU analysis. As shown in Table A-2, μ = 0.6000 and σ = 0.0259. The parameter N is the number of control rods tested at the beginning of the cycle and parameter n is the total number of control rods tested to date for the current cycle. If the cycle average scram time satisfies the Equation A-1 criteria, continued plant operation using the operating limit established with the SCU methodology is permitted. If not, the operating limit for the generator load rejection event must be re-established based on a linear interpolation between the SCU approach and the deterministic approach as follows:

$$OLMCPR_{new} = OLMCPR_{SCU} + T_f\ \Delta OLMCPR \tag{A-5}$$

where

$$T_f \equiv (\tau_{av} - \tau_B)/(\tau_A - \tau_B)$$

τ_A ≡The present technical specification limit on scram time to control rod notch position 39 (0.868 seconds as shown in Table A-2)

ΔOLMCPR is the difference between the operating limit MCPR using the deterministic and the SCU approach

This approach is consistent with the NRC requirements described in Reference A2.

A.7 REFERENCES

A1. Letter from J.M. Sorensen (SLI) to G. Srikantiah (EPRI), "Evaluation of Void Coefficient Uncertainty for WNP-2," August 13, 1991.

A2. 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.

A3. M.A. Alammar, "The NRC Review of Oyster Creek Reload Licensing Model", Proceedings: Sixth International RETRAN Conference, EPRI NP-6949, August 1990.

A4. S.L. Forkner, et al., "BWR Transient Analysis Model Utilizing the RETRAN Program", TVA-TR81-01, Tennessee Valley Authority, 1981.

A5. Letter from R.L. Tedesco (NRC) to G.G. Sherwood (GE), "Acceptance for Referencing General Electric Licensing Topical Report NEDO-24154/NEDE 24154P", February 4, 1981.

Table A-1

Generator Load Rejection Without Bypass
Model Uncertainty Evaluation

	Case*	ΔRCPR
Nuclear Model Uncertainties		
Void Coefficient (13%)	109/110	+0.018
Doppler (-10%)	101/104	+0.005
Prompt Moderator Heating (-25%)	001/105	+0.013
Scram Reactivity (-10%)	001/108	+0.004
Core Thermal Hydraulics Parameters		
Code Correlation (kappa1±0.20)	001/203	+0.001
Code Correlation (CGL±30%)	001/204	+0.003
Code Correlation (CDB±20%)	001/206	+0.001
Code Correlation (CHN±20%)	001/208	+0.001
Core Pressure Loss Coefficients (-20%)	001/212	-0.002
Initial Core Bypass Flow (-20%)	001/213	+0.003
Fuel Pin Radial Nodes (+50%)	001/214	+0.004
Core Power (+4%)	001/216	+0.003
Recirculation System Parameters		
Recirculation Loop Inertia (+100%)	001/301	+0.007
Recirculation Pump Head (-10%)	001/302	+0.003
Jet Pump Inertia (+100%)	001/303	+0.008
Separator Liquid Outlet Inertia (100%)	001/304	+0.001
Jet Pump Loss Coefficient (-20%)	001/305	+0.004
Separator Inlet Inertia (30%)	001/307	+0.003
Steam Line Model Parameters		
Steam Line Inertia (+7%)	001/401	+0.007
Pressure Loss Coefficient (-20%)	001/402	+0.007
Vessel and Loop Geometry Parameters		
Vessel Dome Volume (-5%)	001/501	+0.005
Steam Line Volume (-5%)	001/502	-0.002

Overall Model Uncertainty (ΔRCPR) = 0.0293

* Supply System analysis traceability case numbers. ΔRCPR is the difference in RCPR for the two cases identified.

Table A-2
Scram Times Used in the WNP-2 Safety Analysis

Control Rod Position (ft)	Tech Spec Scram Time (sec)	Normal Scram Time (sec)	Mean Scram Time (sec)*	Standard Dev. (σ) (sec)*	Mean Scram Time+2σ (sec)*
0.0	0.00	0.00	0.00	---	0.00
0.0	0.20**	0.20**	0.20**	---	0.20**
0.75	0.43	0.404	0.3048	0.0161	0.3370
2.25	0.868	0.660	0.6000	0.0259	0.6518
5.75	1.936	1.504	1.3040	0.0633	1.4306
10.75	3.497	2.624	2.3912	0.1354	2.6620
12.00	3.889	2.904	2.6630***	---	2.9698***

* Obtained from WNP-2 scram time surveillance data

** Accounts for 0.20 sec time from de-energization of scram solenoid to the beginning of control rod insertion

*** Extrapolated from scram time surveillance data. Surveillance data is for control rod positions 45 @ 0.75 ft, 39 @ 2.25 ft, 25 @ 5.75 ft, and 05 @ 10.75 ft.

Table A-3

Generator Load Rejection Without Bypass
RCPR As a Function of Scram Time
(Used in Construction of Response Surface)

	ΔCPR	RCPR
Mean Scram Time	0.191	0.160
Mean Scram Time Plus One Standard Deviation	0.174	0.148
Mean Scram Time Minus One Standard Deviation	0.170	0.145
Mean Scram Time Plus Two Standard Deviations	0.224	0.183
Mean Scram Time Minus Two Standard Deviations	0.144	0.126

APPENDIX B

HOT CHANNEL METHODOLOGY

The change in Critical Power Ratio (ΔCPR) during the limiting transient is to be determined in order to establish the minimum CPR operating limit for each reload cycle. The Hot Channel Methodology used by the Supply System for evaluation of ΔCPR is described in Section B.1. This methodology is based on a NRC reviewed core thermal hydraulic code VIPRE-01 MOD02[B1] ("VIPRE-01"). Benchmark studies performed to verify the capability of VIPRE-01 with the ANFB critical power correlation[B2] for ΔCPR calculations are presented in Sections B.2 through B.4.

B.1 VIPRE-01 METHODOLOGY FOR EVALUATION OF ΔCPR

The following discussions describe how VIPRE-01 is used to evaluate the limiting bundle ΔCPR for BWR transients that are analyzed with the RETRAN-02 code. The RETRAN-02 code is used to determine core-wide system response for the limiting transients. The WNP-2 transient analysis model developed with RETRAN-02 is described and qualified in the topical report WPPSS-FTS-129[B3]. In the WNP-2 RETRAN-02 model, the core is modeled with two hydraulic channels: the core average active channel and the bypass channel. These are parallel flow path channels that share common lower plenum and upper plenum boundary conditions on pressure and enthalpy. Modeling of individual fuel bundles is not needed for the purpose of determining core-wide system response. However, because of power level and fuel design differences, a hot channel model is necessary for an accurate evaluation of limiting bundle thermal hydraulic response during a transient. In the hot channel model, the limiting bundle is modeled as a single channel as shown in Figure B-1. This hot channel model developed with VIPRE-01 is used to evaluate ΔCPR.

The Hot Channel Methodology involves four major steps:

1) After the core-wide transient analysis, the boundary conditions for VIPRE-01, namely, the time histories of core power, inlet enthalpy, upper plenum pressure, core pressure drop and the axial power distribution, are saved on a computer file by a RETRAN-02 REEDIT run. This file, after reformatting is read by VIPRE-01.

2) Because the bypass flow is not modeled in VIPRE-01, the lower tie plate loss coefficient is adjusted to give active flow and core pressure drop results identical to RETRAN. A full core VIPRE-01 model with four channels in the active core region is used in this process. The four channels full core nodalization is shown in Figure B-2. The axial nodalization remains unchanged from the hot channel model (See Figure B-1).

3) VIPRE-01 (hot channel model) is run with a successively higher bundle radial peaking factor until the minimum critical heat flux ratio (MCHFR) equals 1.0 at some time during the transient.

4) The bundle power that caused MCHFR = 1.0 is then used to evaluate the initial critical power ratio (ICPR) for the transient. The ΔCPR is equal to (ICPR - 1.0).

Besides RETRAN-02, VIPRE-01 input data are obtained from three other codes: SIMULATE-E[B4], ESCORE[B5] and CASMO-2E[B6].

1) SIMULATE-E is a three dimensional steady-state nodal analysis code which yields the power distributions.

2) ESCORE is a steady-state fuel performance code which yields the gap conductance.

3) CASMO-2E through FICE yields the local peaking function "FEFF". FEFF is needed in the ANFB critical power correlation to account for planar local effects.

B.2 CODE BENCHMARKING

The ANFB critical power correlation developed by Advanced Nuclear Fuels Corporation (ANF) is implemented in the VIPRE-01 code for the thermal margin evaluation. Calculations are performed to benchmark the VIPRE-01 code with steady-state and transient critical heat flux (CHF) test data. Specifically, these calculations provide i) a comparison between critical power results computed with the VIPRE-01 code and steady state critical power test data, ii) VIPRE-01 predictions of transient boiling transition performance versus transient data. This benchmarking therefore provides an overall assessment of how the combination of thermal/hydraulic modeling and CHF modeling in VIPRE-01 performs relative to data for BWR fuel bundle CHF performance. It also verifies the proper implementation of the ANFB correlation.

B.3 BENCHMARKING DATA

The CHF test data to which VIPRE-01 is benchmarked are summarized as follows:

1) Steady-state cases: 96 cases were selected from the ANFB correlation data base[B2]. They are selected from 11 test sections which differ from one another either in terms of geometry (4X4, 5X5 and 9X9 rod configurations) or power distribution (cosine, upskew and uniform axial power distributions)

2) Transient cases: 14 transient tests (7 power ramp tests and 7 flow decay tests[B7]) on a 4x4 rod configuration and 3 transient tests[B8] on a 9x9 rod configuration.

B.4 RESULTS

B.4.1 Steady-state CHF Analysis

A summary of the steady-state CHF analysis results is given in Table B.4-1. Examination of the results shows that VIPRE-01 predicted ECPRs are the same as those reported for the ANFB correlation, with the exception of some small variations between test sections. The overall average difference in ECPR between the VIPRE-01 and ANF's results is 0.0001. It is therefore concluded that the overall mean and standard deviation for the VIPRE-01 predicted ECPR distribution remain essentially unchanged from those reported for the ANFB correlation[B2] (mean = 1.003, standard deviation = 0.025).

B.4.2 Transient CHF Analysis

A summary of the transient CHF analysis results is given in Table B.4-2. For each test case, the initial power was iterated until the predicted time to boiling transition (BT) equaled the measured time to BT. Since the power forcing function (transient bundle power normalized to the initial value versus time) remained unchanged during the iterations, ECPR can be determined as the ratio of predicted initial power to measured initial power (i.e., ECPR = predicted critical power / measured critical power = predicted initial power / measured initial power). The values of ECPR show that the differences between the predicted and measured critical powers are within the uncertainty of ANFB correlation. The uncertainty is incorporated into plant safety limit methodology. These results therefore indicate that the application of the steady-state ANFB correlation in a transient analysis yields CHF results bounding the data (within the uncertainty of ANFB correlation) during the transient. This is consistent with NRC staff's conclusion in the Safety Evaluation Report for VIPRE-01[B8]. The NRC found "... except for very rapid depressurization, the use of CHF correlations developed with steady state CHF data can

correctly or conservatively predict the transient CHF when the instantaneous local fluid conditions are used."

On the basis of the steady-state and transient benchmarks presented here, it is concluded that VIPRE-01 predictions of BWR fuel bundle CHF behavior using ANFB correlation are acceptable for BWR licensing analysis.

TABLE B.4-1

Summary of Steady-state CHF Analysis

Test Section	Run	ECPR[3] (1)	ECPR[3] (2)
JP1	33	1.006	1.008
	46	1.000	1.002
	51	1.007	1.008
	57	1.013	1.014
	62	1.009	1.010
	66	0.996	0.998
	79	1.050	1.051
	91	1.011	1.013
	105	0.942	0.943
	114	0.967	0.969
	116	0.890	0.892
COS	7	1.031	1.029
	15	1.003	1.003
	25	1.020	1.017
	47	0.991	0.989
	52	1.009	1.008
	58	1.034	1.033
	62	1.011	1.011
	81	0.968	0.969
	97	1.000	0.999
UPSKEW	6	1.049	1.048
	33	0.963	0.961
	36	1.000	0.998
	43	0.991	0.989
	57	1.026	1.025
	58	1.028	1.025
	73	0.927	0.926
	84	1.014	1.015
JP7	21	1.017	1.020
	24	1.030	1.033
	33	0.996	0.999
	34	1.040	1.043
	42	1.032	1.034
	44	1.022	1.024
	66	1.020	1.023
	79	0.988	0.991
	80	0.983	0.985
	85	0.979	0.981
	103	0.942	0.945
JP10	8	1.045	1.047
	18	1.025	1.025
	28	1.003	1.005

TABLE B.4-1 (CONT.)
Summary of Steady-state CHF Analysis

Test Section	Run	ECPR[3] (1)	ECPR[3] (2)
JP10	38	1.029	1.029
	48	1.016	1.017
	54	0.994	0.994
	64	1.027	1.029
	74	1.032	1.032
	104	0.976	0.977
	114	0.989	0.990
	120	1.034	1.033
ANFV	14	1.020	1.021
	15	1.016	1.016
	55	1.022	1.020
	56	1.010	1.007
	81	0.954	0.953
	82	0.984	0.983
	107	0.936	0.936
	108	0.951	0.949
ATA62A	8	1.002	1.002
	18	0.984	0.984
	28	0.987	0.986
	38	0.979	0.979
	48	1.029	1.029
	58	1.004	1.004
KWU1A	7	1.039	1.039
	12	1.037	1.038
	17	0.969	0.970
	23	0.966	0.966
	25	0.961	0.961
	26	0.951	0.951
KWU2A	48	1.001	1.000
	58	0.991	0.993
	68	0.986	0.987
	203	0.993	0.994
	213	1.031	1.031
	223	1.056	1.057
ATA714A	10	0.987	0.987
	11	0.984	0.983
	20	1.020	1.020
	21	1.016	1.016
	30	1.010	1.010
	31	1.013	1.013
	40	0.997	0.997

TABLE B.4-1 (CONT.)
Summary of Steady-state CHF Analysis

Test Section	Run	ECPR[3] (1)	ECPR[3] (2)
ATA714A	41	0.995	0.995
	48	0.992	0.991
	49	0.990	0.990
ATA714F	807	0.998	0.997
	808	0.994	0.993
	817	1.025	1.026
	818	1.023	1.023
	827	1.012	1.013
	828	1.022	1.021
	837	0.997	0.997
	846	1.017	1.016
	912	0.963	0.960
	927	0.975	0.975

NOTE:

(1) Predicted by ANF (Ref. 2)
(2) Predicted by VIPRE-01
(3) ECPR = Experimental Critical Power Ratio
= Predicted Critical Power / Measured Critical Power

TABLE B.4-2

Summary of Transient CHF Analysis

4X4 Bundle Power Ramp Test

Run	Initial Power (KW) Measured[2]	Initial Power (KW) Predicted[3]	ECPR[1]
PR001	2010.0	2021.0	1.004
PR002	2008.0	1991.0	0.992
PR003	1888.0	1855.0	0.983
PR004	1891.0	1931.3	1.021
PR005	1659.0	1669.0	1.006
PR006	1658.0	1653.0	0.997
PR007	2011.0	2011.0	1.000
4X4 Bundle Flow Decay Tests			
FD001A	1449.0	1406.0	0.970
FD001B	1452.0	1527.0	1.052
FD002	1449.0	1439.0	0.993
FD003	1455.0	1350.0	0.928
FD004	1476.0	1430.0	0.969
FD005	1730.0	1700.0	0.983
FD006	1735.0	1765.0	1.017

Mean = 0.993
Standard deviation = 0.026

NOTE:

(1) ECPR = Experimental Critical Power ratio
= Predicted Critical Power / Measured Critical Power
(2) Measured Value (Ref. 7)
(3) Predicted by VIPRE-01

B.2 REFERENCES

B1 C.W. Stewart et al.,"VIPRE-01 - A Thermal-Hydraulic Code for Reactor Cores," EPRI NP-254-CCM-A, Revision 3, Volumes I-V, Electric Power Research Institute, August 1989.

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B7 "Results and Analysis of Transient Rod Bundle Boiling Transient Rod Bundle Boiling Transient Tests," XN-NF-487, Exxon Nuclear Company, October 1982.

B8 "Safety Evaluation Report on EPRI NP-2511-CCM VIPRE-01," NRC, May 1986.

UTP - Upper Tie Plate
TAF - Top of Active Fuel
SPA - Spacer
BAF - Bottom of Active Fuel
LTP - Lower Tie Plate
ORIF - Orifice

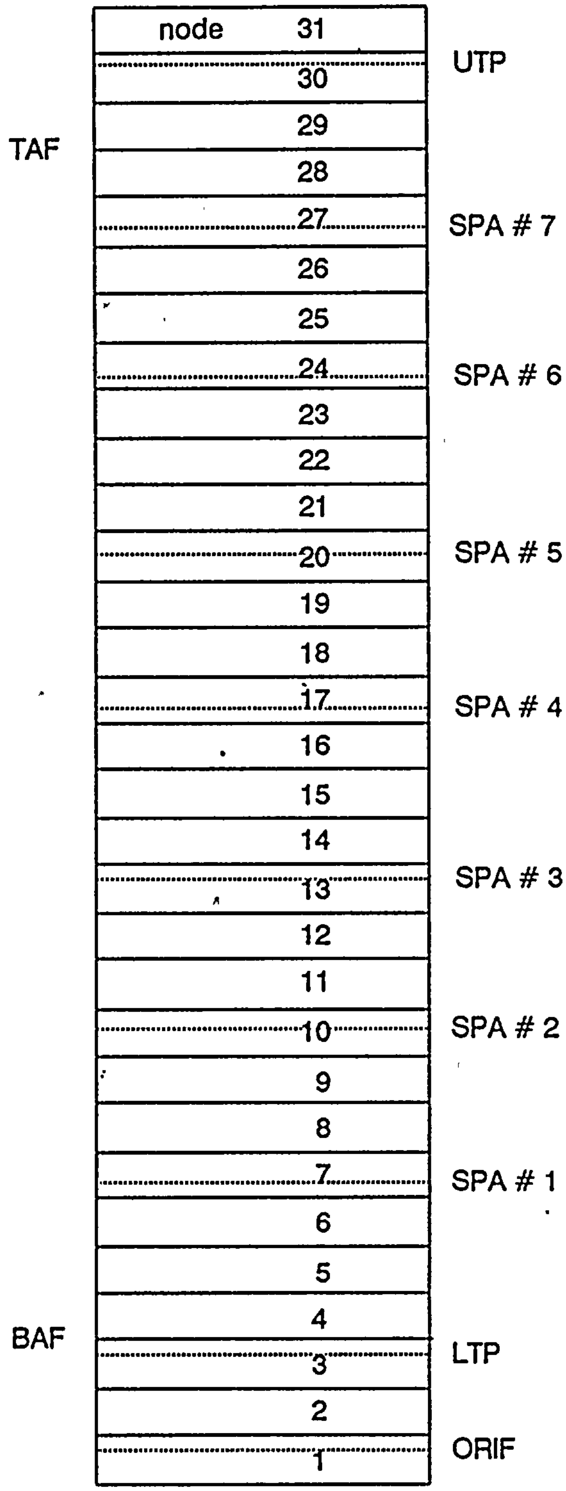
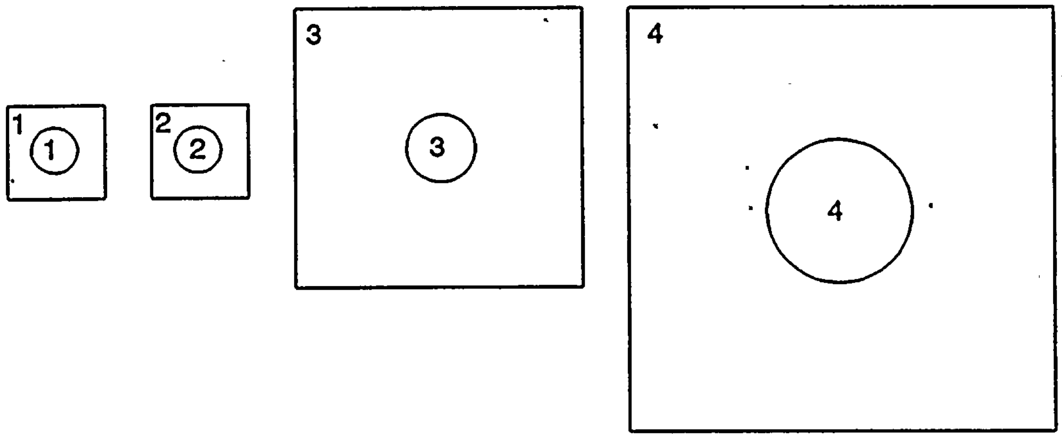


Figure B-1. WNP-2 VIPRE Hot Channel Model



Channel 1 models one bundle, the hottest.peripheral bundle
Channel 2 models one bundle, the hottest central bundle
Channel 3 models 91 peripheral bundles
Channel 4 models 671 central bundles
Rod 1 models 62 fuel rods
Rod 2 models 62 fuel rods
Rod 3 models 91 x 62 = 5,642 fuel rods
Rod 4 models 671 x 62 = 41,602 fuel rods

Figure B-2. Four Channels Full Core Nodalization

APPENDIX C
GENERATION OF KINETICS DATA

The one-dimensional kinetics data for RETRAN-02 described herein uses nuclear cross-sections information prepared by the core analysis methodology described in Reference C1. Files containing kinetics data for the various fuel types present in the reactor core are produced by CASMO-2E and the three-dimensional nodal characteristics of the core are determined by SIMULATE-E. SIMTRAN-E collapses corewide cross-sections from the three-dimensional form to one-dimensional form as required by RETRAN-02. SIMULATE-E and RETRAN-02 calculate the moderator density differently. Because of this, the SIMTRAN-E cross-sections are adjusted to account for the difference. To do this adjustment, there are additional steps required in the SIMTRAN-E to RETRAN-02 one-dimensional kinetics file transfer. These steps, which involve completion of additional SIMULATE-E and RETRAN-02 cases and the execution of the STRODE code, are required to adjust the moderator density coefficients. STRODE also performs an adjustment of the delayed neutron fraction.

C.1 GENERAL DESCRIPTION OF THE GENERATION OF ONE-DIMENSIONAL DATA

The generation of the initial one-dimensional data file for RETRAN-02 uses the EPRI codes SIMULATE-E and SIMTRAN-E. SIMULATE-E predicts core power and burnup distributions during detailed depletion analyses of the reactor core. Qualification of the Supply System's SIMULATE-E methodology is provided in Reference C1.

SIMTRAN-E was developed under EPRI sponsorship for linking SIMULATE-E and RETRAN-02. SIMTRAN-E reads restart files written by SIMULATE-E, extracts the appropriate information for determining the kinetics parameters, and generates the direct RETRAN-02 input for transient analysis.

SIMTRAN-E produces a one-dimensional kinetics data file in the form of polynomials which describe the effects of relative changes in water density and fuel temperature on calculated two-group cross-sections, diffusion coefficients, inverse velocities, radial bucklings, and delayed neutron fractions. This data file could be used directly by RETRAN-02. However, without the adjustments described below and the use of the STRODE code, this approach would lead to very conservative RETRAN-02 predictions for severe pressurization events.

STRODE uses the results of a set of several SIMULATE-E and RETRAN-02 cases to quantify the effects of the difference in axial moderator density distributions between the two codes for identical variations in core pressure. The differences between the axial arrays produced by SIMULATE-E and RETRAN-02 are used by STRODE to adjust the moderator density coefficients from SIMTRAN-E to obtain consistent moderator density reactivity feedback between SIMULATE-E and RETRAN-02. The STRODE output data is in the same form as that produced by SIMTRAN-E and is directly used as RETRAN-02 input.

STRODE is also used to correct the delayed neutron fractions in the cross-section libraries used by CASMO-2E and processed through NORGE-B. The CASMO-2E cross-section library (ENDF/B-III) delayed neutron fractions are lower than that provided in the more recent cross-section library (ENDF/B-V). STRODE is used to adjust the delayed neutron fraction input to RETRAN-02 to be consistent with the more recent data.

C.2 CALCULATION OF THE INITIAL ONE-DIMENSIONAL DATA

To generate the initial one-dimensional data, a base SIMULATE-E case [shown as (1) in Figure C-1] is run at a core configuration consistent with the initial conditions for the given transient. This base SIMULATE-E case uses power and void feedback to determine the three-dimensional core power and flux distributions and the

critical eigenvalue. The base SIMULATE-E case uses cross-sections which have not been adjusted to match k-infinities of the lattice physics code (i.e. unadjusted $\Sigma a1$), but it is run from a SIMULATE-E restart file generated with cross-sections which were adjusted to match k-infinities of the lattice physics code (i.e. adjusted $\Sigma a1$).

For transients in which it is necessary to model the effects of control rod insertion resulting from a scram, an additional SIMULATE-E case is required [shown as (2) in Figure C-1]. This case is based on the nominal case and is run with power feedback disabled. The only difference between this case and the nominal case is the control rod position array, which has all rods fully inserted.

As noted above, SIMTRAN-E [shown as (3) in Figure C-1] reads the restart files generated by the SIMULATE-E cases. It then collapses the three-dimensional SIMULATE-E data to one-dimensional data for RETRAN-02 and determines the dependence of the kinetics parameters on relative water density, square root of fuel temperature, and control state.

SIMTRAN-E generates all kinetics parameters except $k\Sigma f1$ and $k\Sigma f2$ by radial collapse with adjoint flux weighting. $k\Sigma f1$ and $k\Sigma f2$ are radially collapsed with volume weighting. The dependence of the kinetics parameters on water density and fuel temperature is determined by making perturbations in these quantities. All relative water density and square root of average fuel temperature perturbations are done in three dimensions and are then radially collapsed. The one-dimensional nominal and perturbed kinetics variables are then analyzed to produce polynomials that are functions of the relative change in water density and the change in the square root of the average fuel temperature at each axial node. This procedure is performed for both the nominal case and for the controlled case.

C.3 ADJUSTMENT OF KINETICS DATA

To begin the second step in the generation of the one-dimensional data, the SIMTRAN-E output from step (3) of Figure C-1 (called the Unmodified 1-D File) is used and parallel SIMULATE-E and RETRAN-02 cases are run to quantify the difference in axial moderator density distributions between the two models for identical variations in core pressure. These runs are shown as steps (4), (5) and (6) in Figure C-1.

The extra SIMULATE-E delta pressure cases in step (6) are used as moderator density perturbation sets for each axial node of the one-dimensional core model for the final SIMTRAN-E run shown in step (9) of Figure C-1. An extra SIMULATE-E calculation is performed at 10% higher power than the base case [shown in step (7) of Figure C-1]. A special SIMTRAN-E calculation is performed to determine the one-dimensional temperature distribution for this increased power case [shown in step (8) of Figure C-1]. The differences in temperature from the step (3) case and the step (8) case are used as a fuel temperature perturbation set for each axial node of the one-dimensional core model in the final SIMTRAN-E run [step (9)]. A final SIMTRAN-E run is then performed with these perturbation sets to generate a one-dimensional data file for STRODE.

The coefficients in the kinetics parameter polynomials that are associated with the changes in relative moderator density are then modified by STRODE [step (10) of Figure C-1] so the change in each kinetics variable in RETRAN-02 for a given pressure change is the same as that pressure change would give in the one-dimensional SIMTRAN-E model. Thus reactivity changes at each axial node generated by a pressure change are preserved between SIMTRAN-E and RETRAN-02. The constant term in each kinetics parameter polynomial determines the initial steady state eigenvalue in the RETRAN-02 unperturbed state. The constant terms are not modified when new

polynomial fits are developed; consequently, the SIMULATE-E eigenvalue is preserved in the RETRAN-02 unperturbed state.

The cross-section libraries used in the core physics analysis are based on ENDF/B-III. ENDF/B-III includes delayed neutron fractions which are artificially low. Preliminary ENDF/B-V data shows an increase in delayed neutron fraction ranging upwards from 5.4% in all fissile isotopes. To bring the delayed neutron fraction closer to those specified in ENDF/B-V, a +5% adjustment is applied in the STRODE code to all delayed neutron fractions before final data is put into the RETRAN-02 input file.

C.4 VERIFICATION OF THE SUPPLY SYSTEM'S METHODOLOGY

Verification of the Supply System's methodology is performed in Reference C2. In this reference, a comparison in made between the axial power distributions produced by SIMTRAN-E to the initial states of the three Peach Bottom turbine trip tests. Also in Reference C2 is a comparison of results calculated using the Supply System's methodology to the measured results of the Peach Bottom turbine trip tests. In these results the calculated response of the Supply System's model closely matches the measured results.

C.5 REFERENCES

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C2. Y.Y. Yung, S.H. Bian, and D.E. Bush, "BWR Transient Analysis Model," WPPSS-FTS-129, Rev. 01, Washington Public Power Supply system, September 1990.

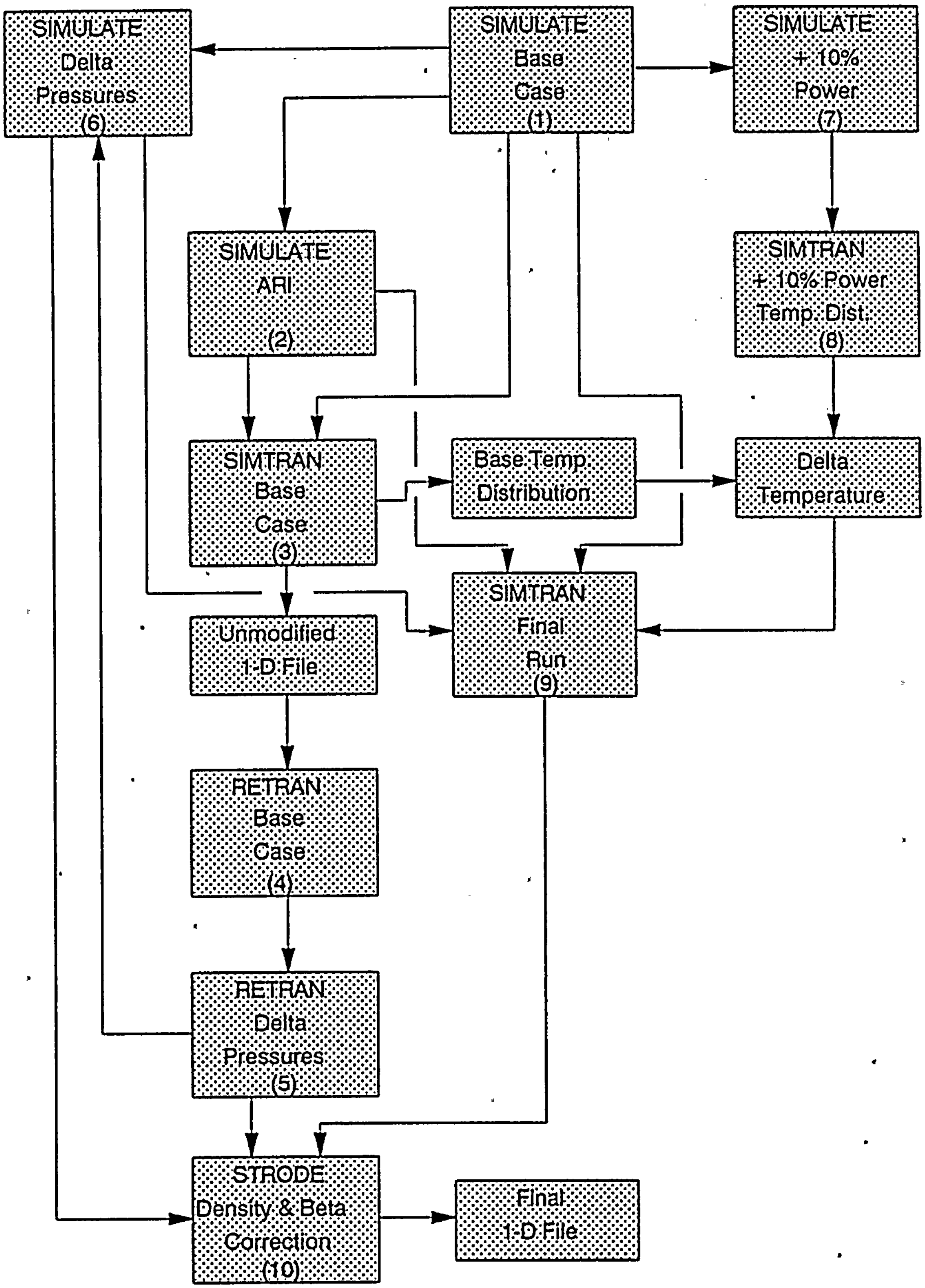

Figure C-1. Generation of Kinetics Data

APPENDIX D
GAP CONDUCTANCE CALCULATION

This appendix presents the calculations of the fuel-cladding gap conductances using the ESCORE code. Gap conductances are needed in modeling the thermal behavior of the fuel pellet and cladding and the heat transfer to the coolant during a simulation of operational and other transients. In the thermal margin calculations, two different types of gap conductances are required: core-average and hot channel. The core-average gap conductance is used in the RETRAN-02 simulation of a transient. The hot channel gap conductance is used in the VIPRE-01 hot channel CPR analysis. They are discussed separately in the following sections. The ESCORE code, as approved by the NRC, is described in Section 2.1.4.1.

D.1 CORE-AVERAGE GAP CONDUCTANCE

D.1.1 General Approach

The approach used for the core-average calculation is to separate the fuels in the reload core to be analyzed into different "batches", where batch is defined as any group of fuel assemblies that are introduced into the core at the same time with common mechanical and nuclear designs. For Cycle 4 core, there are 6 batches ranging from Batch 2 through Batch 7 (Batch 1 was discharged in Cycle 3). The Cycle 4 configuration is selected to demonstrate the methodology of calculating the gap conductances. The approach can be applied to any other cycles with appropriate changes in fuel design parameters and operating conditions and histories. An input deck for each batch through its life in the core is prepared using: the radial power, axial power, exposure and axial relative moderator density from neutronic calculations; the fuel dimensions, enrichment, fill gas type and pressure, and surface roughness from the vendor drawings and documents; and coolant conditions from the FSAR and RETRAN-02 calculations. The gap conductance for each batch at the end of Cycle 4 (EOC4) is then calculated using the ESCORE code.

The calculated batch gap conductances are averaged by an algorithm which preserves the core-wide heat transfer from the fuel pellets to the cladding.

D.1.2 WNP-2 ESCORE Model Description

The ESCORE calculation starts with the input preparation for all fuel batches in the Cycle 4 core. The size of the batches and the average enrichments are given in Table D-1. The fuel dimensions and other design parameters are obtained from the fuel vendor (ANF) and the power and void histories are obtained from neutronic calculations. Since the Cycle 4 reload is simulated in the middle of Cycle 3 operation, the actual power and burnup histories for each batch is only available for Cycles 1 and 2. These parameters for Cycles 3 and 4 must be estimated based on predicted core loading as dictated by cycle energy requirements. The detailed fuel behavior (i.e., pellet and cladding temperatures, fission gas release, cracking, densification, swelling, clad creepdown and gap conductance) for each batch is calculated by the ESCORE code from the time the batch is loaded in the core to the EOC4. At EOC4, an extra time step of 0.1 days is allowed to simulate the licensing conditions (at 3468 MWt), instead of the nominal operating conditions, as required for the licensing transient analysis.

Table D-1

Batch No.	Batch Size	Fuel Type	Cycle Loaded	Fuel Enrichment* (w/o U-235)
2	56	GE 8x8R	1	1.853
3	280	GE 8x8R	1	2.321
4	128	ANF 8x8	2	2.89
5	148	ANF 8x8	3	2.89
6	24	ANF 8x8	4	2.89
7	128	ANF 8x8	4	2.81

* The enrichment is for the center portion. There are two reflector regions, each 6 inches long, at the top and bottom of each fuel rod that are filled with natural uranium oxide.

As seen from Table D-1, there are 336 GE 8x8R fuel bundles in the Cycle 4 core. However, since the GE fuel is located in lower power zones, and the ANF 8x8 fuel design is very similar to the GE fuel in the Cycle 4 core, the GE fuel was simulated as ANF 8x8 fuel.

There are 24 axial nodes in the ESCORE model. The ESCORE code calculates the axial gap conductances at each burnup step. As mentioned earlier, a small time step of 0.1 days is added at the EOC4 to calculate the gap conductances at the licensing condition of 104.4% power.

The batch axial gap conductances thus obtained are collapsed into a single value for the batch averaged gap conductances, which are further collapsed into the core-averaged gap conductance. An algorithm was developed for this collapsing process and is summarized below.

When plant operational transients are analyzed with the RETRAN-02 code, the core is modelled as a one-dimensional (axial) region, and the rate of heat transfer across the gap between the fuel pellets and the cladding is characterized by a single value of gap

conductance. As stated in Reference 9, the use of a constant gap conductance is conservative for system analysis involving power increases such as pressurization transients. The gap conductance required by the RETRAN-02 input is used to calculate the heat transfer from the pellet to the cladding which affects the fuel temperature, stored energy, and coolant thermal-hydraulics. Therefore, the averaging procedure should preserve the total rate of heat transfer from the fuel to the cladding. This requirement leads to a definition of average gap conductance in which nodal values of gap conductance are weighted by the product of the nodal heat transfer area and nodal fuel-cladding temperature difference. The resulting core average gap conductance is given as:

$$\langle h_g \rangle = \frac{\sum_b N_b \langle h_{gb} \rangle \langle T_{fb} - T_{cb} \rangle}{N \langle T_f - T_c \rangle} \qquad \text{(Eq. D.1)}$$

where <> represents averaged quantities and

$$\langle h_{gb} \rangle = \frac{\sum_i \Delta z_{ib} h_{gib} (T_{fib} - T_{cib})}{L \langle T_{fb} - T_{cb} \rangle} \qquad \text{(Eq. D.2)}$$

$$\langle T_f - T_c \rangle = \frac{\sum_b N_b \langle T_{fb} - T_{cb} \rangle}{N}$$

$$\langle T_{fb} - T_{cb} \rangle = \frac{\sum_i \Delta z_{ib} (T_{fib} - T_{cib})}{L}$$

where

Δz_{ib} = height of ith axial node in batch b

$L = \sum_i \Delta z_{ib}$

N_b = number of assemblies in batch b

$N = \sum_b N_b = 764$ for WNP-2 plant

h_g = gap conductance

T_f = pellet surface temperature

T_c = cladding inner surface temperature

Subscript

i = axial node (1,2,3...,24)

b = batch number

The batch average gap conductances are first calculated based on Equation D.2. The results for the 6 batches at the EOC4 licensing condition are given in Table D-2:

Table D-2

Batch Average Gap Conductance

Batch No.	(Btu/hr/ft/ft/F)
2	566.8
3	751.8
4	912.5
5	921.6
6	973.7
7	903.6

These batch average gap conductances are combined using Equation D.1 to give a value of 839.9 Btu/hr/ft^2/^{o}F for the core average gap conductance. This core average value, after multiplying by the gap width, is used in the RETRAN-02 transient analysis at the 104.4% power licensing condition for WNP-2 at EOC4.

A burnup time step of 40 days is used in the ESCORE model. A sensitivity study was made by reducing the burnup time step to 20 days. The results of the sensitivity study are essentially identical to the base model in terms of the average gap conductance. Therefore, the step size of 40 days is used for all of the calculations.

D.2 HOT CHANNEL GAP CONDUCTANCE

D.2.1 General Description

The hot channel gap conductance is obtained by first selecting the fuel type that is expected to be the hot channel and then ramping the power to the high radial peak value that will be experienced by the hot channel. The subsequently obtained gap conductance is used in the VIPRE-01 hot channel CPR analysis.

The methodology used is the same as that used for the calculation of the core average gap conductance. The only difference between this calculation and the core average calculation is that only the fresh fuel (which is exposed to the highest radial peaking) is considered in this analysis and the power level during the last time step for the batch under consideration is adjusted to give the desired hot channel radial peaking.

D.2.2 Analysis Results

This section contains the results of the ESCORE runs made to determine the hot channel gap conductance. As mentioned previously, the calculations were carried out by modifying the power of the hot bundle in the last time step.

Seven ESCORE runs were executed for this analysis. Six runs were made using fuel batch 7 with radial peaking factors (RPF) between 1.3 and 2.1 (RPF and APF of 1.0 equates to an APLHGR of 5.68 kW/ft for the licensing core power of 3468 MWth). An additional case was

run for batch 6 at a RPF of 1.5. The results of the ESCORE runs were used to determine the bundle average gap conductance for the hot bundle.

ESCORE generates nodal gap conductance values for each axial node in the hot bundle. As in the core-average case, these nodal values must be collapsed to give one value that is appropriate for the hot channel. Equation D.2 in Section D.1.2 was used for this collapsing. Results after collapsing for each of the seven runs are given in Table D-3.

Table D-3

ESCORE Hot Channel Results

Fuel Batch	Flow Rate	Peaking Factor	Peak Bundle APLHGR (kw/ft)	Hot Channel Gap Cond. (Btu/hr/ft/ft/F)
7	Nominal	1.3	9.31	989.8
7	Nominal	1.4	10.03	1088.0
7	Nominal	1.5	10.74	1205.4
7	Nominal	1.7	12.18	1488.8
7	Nominal	1.9	13.61	1821.9
7	Nominal	2.1	15.04	2182.0
6	Nominal	1.5	10.74	1218.8

D.2.3 Discussion of Results

The results compiled in Table D-3 for batch 7 are plotted on Figure D-1 for RPF's ranging between 1.3 and 2.1. From the plot it can be seen that the gap conductance is non-linear between RPF's of 1.3 and 2.1. This non-linearity indicates that as the RPF is increased, the transient ΔCPR will be increased a higher relative amount than the increase in RPF. In other words, a 10% increase in RPF might produce a higher than 10% increase in transient ΔCPR. It is thus important to use a conservative gap conductance that is derived from an RPF that is appropriate for the fuel in the core but not excessively high.

The hot channel transient methodology employed in the Supply System analysis uses a single value of hot channel gap conductance. This requires that the gap conductance be selected so that it is conservative for the reactor conditions at the initiation of the worst possible transient.

In general, the bundle LHGR and CPR must be within the licensed limits. In the hot channel transient model, the bundle power is iteratively increased at the beginning of the transient until, at the limiting point in the transient, a CPR of 1.0 is reached. During a generator load rejection without bypass transient, this power typically occurs when the APLHGR is about 10 to 11.5 kW/ft for ANF 8x8-2 fuel. The hot channel power is artificially high due to the fact that the CPR is being pushed to 1.0 rather than the safety limit value (i.e., 1.06). It is therefore conservative to pick a gap conductance value that is derived at an APLHGR that bounds the APLHGR reached in the transient calculation (APLHGR equals the bundle power in VIPRE-01 adjusted for direct bypass heating divided by the number of rods in the assembly divided by the fuel assembly length times the APF). From Table D-3, the APLHGR for a RPF of 1.5 is 10.74 Kw/ft. This value should be valid for the limiting licensing load rejection transient for Cycle 4. In the transient methodology it should be pointed out that the APLHGR of the hot bundle at the beginning of the transient should be less than or equal to the APLHGR used in determining the gap conductance. If the APLHGR is greater than that used in the gap conductance calculation, the analysis may still be valid as long as the APLHGR required to get the bundle to the safety limit in the transient is less than the APLHGR used in the gap conductance calculation.

As mentioned earlier, the power history used in this analysis was identical to the power history used to calculate the core average gap conductance for the batch under consideration except for the final time step. This approach is similar to that of the fuel vendor (ANF). ANF selects the hottest bundle at EOC and uses its power history through the cycle in the hot channel gap conductance

determination. The ANF approach is similar to the Supply System method because the hot bundle follows the batch average power history fairly closely through the cycle.

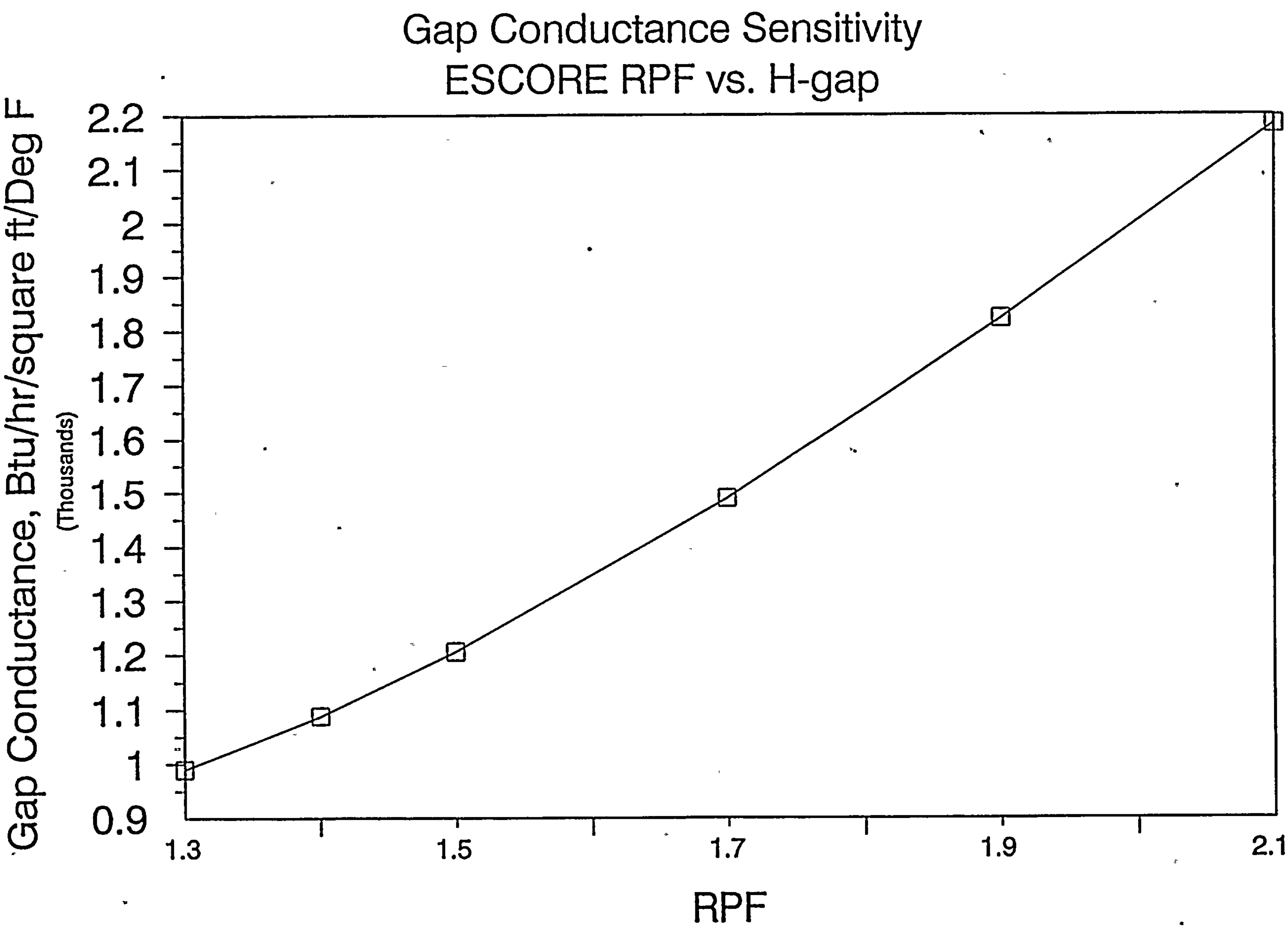

Figure D-1. GAP CONDUCTANCE VS. BUNDLE PEAKING

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ENCLOSURE 1

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
AUGUST 28, 1991
MEETING ATTENDEES

Nuclear Regulatory Commission

W.T. Russell
B.A. Boger
A.C. Thadani
G.M. Holahan
T.R. Quay
R.C. Jones, Jr.
L.F. Miller (RV)
C.A. VanDenburgh
A.E. Cubbage
P.L. Eng
R.K. Frahm, Sr.
J.D. Monninger
J.F. Munro
T. Sundsmo
T.E. Walker (RI)

Washington Public Power Supply System

A.L. Oxsen, Deputy Managing Director
A.G. Hosler, WNP-2 Licensing Manager
S.R. Kirkendall, Supervisor, Nuclear Engineering
L.D. Sharp, Principal Nuclear/Mechanical Engineer
R.W. Conserriere, Operations Shift Manager

Bonneville Power Administration

D.L. Williams, Nuclear Engineer

ENCLOSURE 2

DEVIATIONS RELATED TO NRC CONCERNS

Deviations which are examples of the four identified concerns are noted after each concern by deviation number. This tabulation is not all inclusive.

CONCERN 1:

The quality and depth of effort associated with identification and justification of deviations from the BWROG EPGs was not adequate. The documentation of the justifications did not provide sufficient information to evaluate the technical adequacy of the deviation in many cases.

Deviations: I-4, I-26, I-28, I-32, I-35, D-7, D-10, D-15, S-8

CONCERN 2:

Application of the licensing design basis analysis was inappropriately applied when identifying and justifying deviations. Conclusions were based on a design basis analysis for specific accident sequences, excluding consideration of other malfunctions or adverse conditions.

Deviations: I-28, D-15, S-8

CONCERN 3:

Deviations were taken that removed available equipment and mitigation strategies for use based on operator judgement without sufficient analysis of the safety significance of removing the option.

Deviations: D-4, D-6, S-3

CONCERN 4:

Some PSTG steps and EOP flowcharts do not reflect the accident strategy described in the deviation documentation.

Deviations: D-10, D-15, S-6

WNP-2
EOP DEVIATION PRESENTATION
MEETING AGENDA

I.	INTRODUCTION OF PRINCIPALS	5 minutes
II.	PRESENTATION ORIENTATION	20 minutes
III.	OVERVIEW OF WNP-2 DEVIATIONS	2 hours

A. Design Deviations
B. Strategy Deviations

IV. DETAILED TECHNICAL DISCUSSION ON EACH DEVIATION — 4 hours

- Selected Implementation Deviations

V. CLOSING COMMENTS

SUPPLY SYSTEM PERSONNEL

A. L. Oxsen	Deputy Managing Director
C. M. Powers	Director of Engineering
A. G. Hosler	Manager, WNP-2 Licensing
S. R. Kirkendall	Supervisor, Nuclear Engineering
L. D. Sharp	Principle Nuclear Engineer
R. W. Conserriere	Operations Shift Manager (EOP Coordinator)

ENCLOSURE 3

WNP-2 EOP DEVIATION PRESENTATION

I. INTRODUCTION

A. Meeting agenda and Supply System representation

B. WNP-2 has had considerable involvement in BWROG EPG development.

C. We believed we could resolve design basis conflicts with EPG strategies through a deviation process with adequate justifications.

D. Irrespective of the design basis conflicts, our proposed actions provide technically correct actions.

E. Purpose of today's presentation is to provide detailed information on our deviation justifications.

II. EOP DEVELOPMENT GUIDELINES

A. BWROG'S EPG, Rev. 4, is optimum strategy for severe accident mitigation.

B. Our EOPs should specify additional operator actions to deal with accidents within design basis consistent with our design basis.

C. EPG strategy will be employed if accident degrade beyond design basis.

D. EOPs should accomodate our Licensing Basis where overlap exists with EPGs.

E. Our EOP deviations are not alternative strategies at the exclusion of the EPGs.

F. To the extent possible, we should employ the strength of our integrate[illegible] WR 5, Mark II containment designs to simplify operator actions and/or provide improved accident mitigation.

G. Credibility of our EOP strategies and operator training on the EOP bases is vital to crew performance.

H. Human factors considerations should be factored into our formulation of strategy implementation.

III. EOP DEVIATION CHARACTERIZATION

A. Design deviations

B. Strategy deviations

C. Implementation deviations

D. Present status:

11	Design Dev.
6	Strategy Dev.
28	Implementation

IV. IMPLICATIONS OF OUR EOP APPROACH

A. Successfully resolves the design bases/EPG conflicts without licensing bases changes.

B. Preserves continuity of design bases integration into Operations and Training.

C. Maintains compliance with EPG Safety Evaluation Report.

D. For those positions to be modified, we have provided bases to allow interim operation while EOP bases revision process proceeds.

V. CONTENT OF OUR PRESENTATION PACKAGE

A. Format: Description of deviation, reason, and technical bases for deviation, impact on EPG strategy, and safety significance assessment.

B. Approach: Deviations intended to provide additional operator guidance (hold points) to determine if design was successful; then implement severe accident mitigation strategy if required.

C. Assessment of deviation impacts on severe accident mitigation.

1. Licensing Basis envelope of event sequences identified on EOPs.

2. Reviewed Licensing Basis event consequences with EPG strategies.

3. Where no unacceptable consequences were created by EPG actions, EPG direction was retained and documented by 50.59 process.

4. Where preservation of Licensing Basis was required to maintain License limits, EPG strategy was modified to accommodate License Basis.

5. This approach resolves EPG/Licensing Basis conflicts with minimal impact on Severe Accident mitigation.

DESIGN DEVIATIONS - 12 TOTAL

APPROACH

- SAFETY SIGNIFICANCE EVALUATION PERFORMED FOR AREAS OF DIFFERENCE BETWEEN EPG AND LICENSING ANALYSIS.

- WHERE EPG'S ADDED NO UNACCEPTABLE CONSEQUENCES WITHIN LICENSING ENVELOP, EPG STRATEGY RETAINED, IMPACT EVALUATED VIA 50.59.

- WHERE PRESERVATION OF LICENSING BASIS WAS REQUIRED TO MAINTAIN LICENSING LIMITS, EPG STRATEGY SUPPLEMENTED TO ACCOMMODATE LICENSING BASIS

DESIGN DEVIATION #1

SEPARATE ECCS PUMP NPSH LIMITS TO PREVENT PUMP DAMAGE ARE NOT APPLIED AT WNP-2.

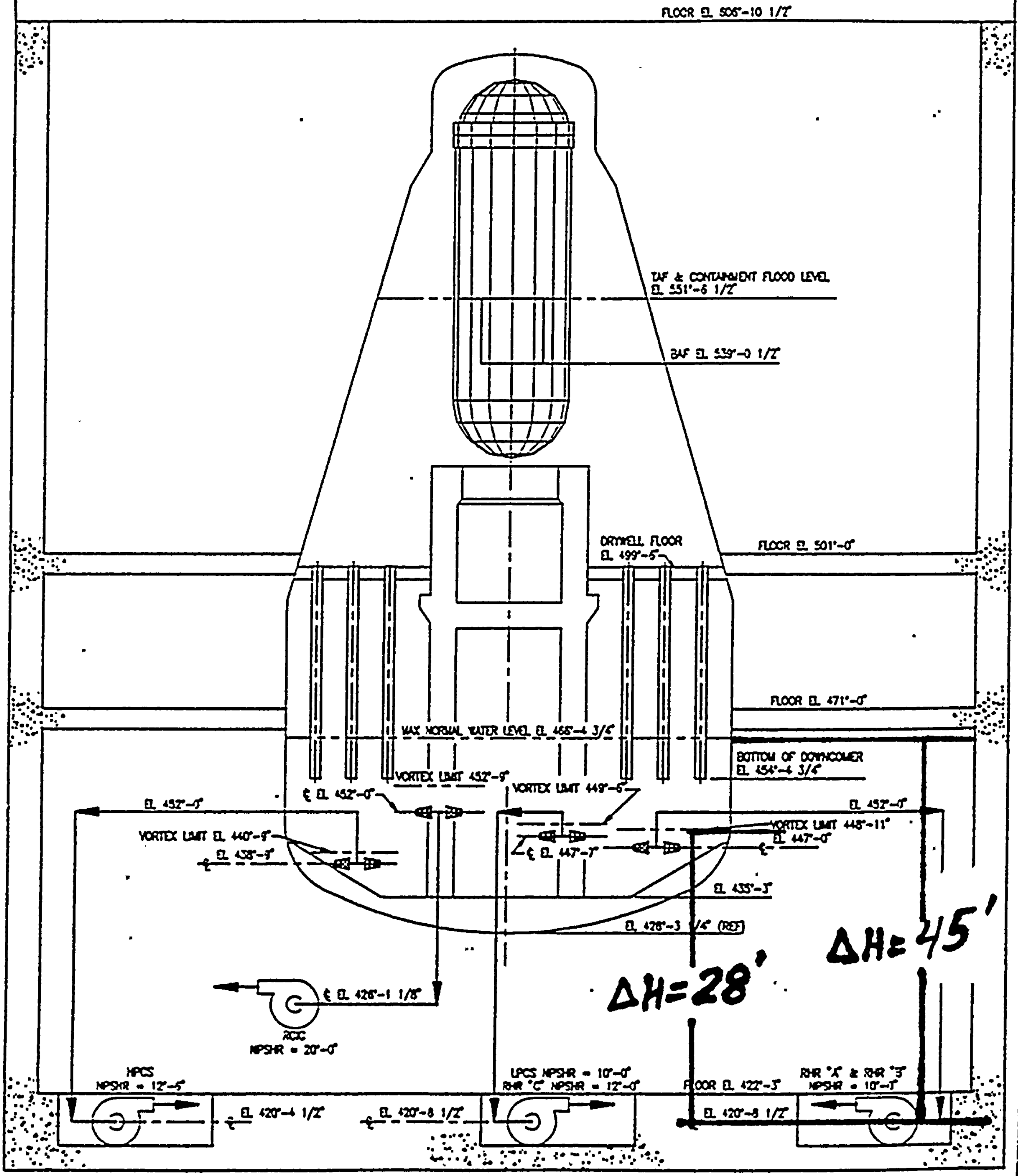
DEVIATION BASIS

- THE NPSHR FOR THE WNP-2 ECCS (INCLUDING RCIC) PUMPS IS ALWAYS LESS THAN THE NPSHA FOR THESE PUMPS AT ANY PUMP FLOW OR WETWELL TEMPERATURE CONDITION WHEN SUPPRESSION POOL LEVEL IS AT OR ABOVE THE VORTEX LIMITS OF THE RESPECTIVE PUMP.
- VORTEX LIMITS BOUND THE NPSH LIMITS AND WOULD ALWAYS BE INVOKED BEFORE NPSH BECOMES LIMITING.

SAFETY SIGNIFICANCE

- THE FULL INTENT OF THE BWROG EPG IS IMPLEMENTED.
- APPLYING VORTEX LIMITS PROTECTS PUMPS FOR BOTH NPSH AND VORTEX CONCERNS

NET POSITIVE SUCTION HEAD FOR
WNP-2 ECCS PUMPS



DESIGN DEVIATION #3

A CAUTION HAS BEEN ADDED TO ALL STEPS WHICH REQUIRE THE USE OF DRYWELL SPRAYS TO ALERT THE OPERATORS TO THE POTENTIAL FOR EXCEEDING THE CONTAINMENT NEGATIVE PRESSURE LIMIT.

DEVIATION BASIS

WNP-2 IS THE ONLY DOMESTIC BWR WHOSE CONTAINMENT DESIGN INCORPORATES A FREE STANDING STEEL SHELL. ANALYSES SHOW THAT SIMULTANEOUS INITIATION OF A DRYWELL SPRAY AND ANY OTHER CONTAINMENT SPRAY IN CONJUNCTION WITH A SINGLE WETWELL TO REACTOR BUILDING VACUUM BREAKER OUT OF SERVICE HAS THE POTENTIAL TO FAIL CONTAINMENT DUE TO NEGATIVE PRESSURIZATION.

SAFETY SIGNIFICANCE

DEVIATION DOES NOT ALTER BWROG STRATEGY. WNP-2 SPRAYS ARE UTILIZED FOR THE SAME PURPOSES UNDER THE SAME CONDITIONS AND AT THE SAME POINTS AS REQUIRED IN THE GENERIC.

SAFETY SIGNIFICANCE OF NOT ADDING THE CAUTION COULD BE THE LOSS OF CONTAINMENT INTEGRITY AND RESULT IN AN UNCONTROLLED RELEASE OF RADIOACTIVE CONTAMINANTS TO THE ENVIRONMENT.

DESIGN DEVIATION #4

WHEN USING TURBINE BYPASS VALVES TO DEPRESSURIZE RX, (NON-ATWS ONLY) OPERATORS ARE NOT ALLOWED TO EXCEED TS COOLDOWN RATE UNTIL EMERG DEPRESS IS REQ'D. EPG'S DIRECT OPERATORS TO RAPIDLY DEPRESS THE RX EXCEEDING COOLDOWN RATE IF EMERG DEPRESS IS "ANTICIPATED".

DEVIATION BASIS

- IT IS VERY DIFFICULT TO PROVIDE CLEAR GUIDANCE TO THE OPERATORS AS TO WHEN "EMERGENCY DEPRESSURIZATION IS ANTICIPATED" OR HOW FAST "RAPID DEPRESSURIZATION" IS AS CALLED FOR BY EPG STRATEGY

- BWROG EPG'S DO NOT DEFINE "ANTICIPATE" OR "RAPID" DEPRESSURIZATION. NO GUIDANCE PROVIDED. ONLY PLACE IN EOP'S OPERATOR IS DIRECTED TO ANTICIPATE.

- LACK OF CLEAR OPERATOR GUIDANCE RESULTS IN INCONSISTENT OPERATOR RESPONSES AND COULD CAUSE UNNECESSARY AND SEVERE RX THERMAL/PRESSURE FATIGUE CYCLES.

- WNP-2 BELIEVES (CONSISTENT WITH EPG REV. 4 SER) THAT ALLOWANCE TO EXCEED LICENSING LIMITS SHOULD ONLY BE PROVIDED WHEN PLANT CONDITIONS HAVE DEGRADED BEYOND LICENSING BASES/SPECIFICATIONS. PERMISSION TO EXCEED COOLDOWN AT THIS EARLY POINT IN THE STRATEGY COULD RESULT IN EXCESSIVE COOLDOWN WHILE WITHIN LICENSING BASIS.

SAFETY SIGNIFICANCE

AT THIS EARLY STAGE IN THE STRATEGY, EPG'S DO NOT ALLOW OVERRIDE OF ANY ISOLATION INTERLOCKS. CONSEQUENTLY, SITUATIONS WHERE EMERG DEPRESS CAN BE "ANTICIPATED" AND MSIV'S ARE OPEN AND THIS PATH HAS NOT BEEN LOST, ARE VERY DIFFICULT TO IDENTIFY, BUT ARE BELIEVED TO BE A VERY SMALL FAMILY OF EVENTS.

THIRTEEN CONDITIONS (NON-ATWS) WHICH ARE CLEARLY DEFINED, REQUIRE EMERGENCY DEPRESSURIZATION. WHEN THE RX MUST BE EMERG. DEPRESS. THE OPERATOR IS DIRECTED TO TAKE THE NECESSARY ACTIONS WHILE CONDITIONS ARE STILL WELL WITHIN THE HEAT CAPACITY LIMITS OF THE PLANT.

ANALYSES DEMONSTRATE THAT WNP-2'S LARGE SP VOLUME (840,000 GAL) PROVIDES THE HEAT CAPACITY TO ACCOMMODATE A WIDE SPECTRUM OF ACCIDENTS WITHOUT SIGNIFICANT CHALLENGE TO CONTAINMENT LIMITS.

WNP-2 DOES NOT PRECLUDE PREFERENTIAL USE OF BYPASS VALVES TO DEPOSIT ENERGY OUTSIDE CONTAINMENT.

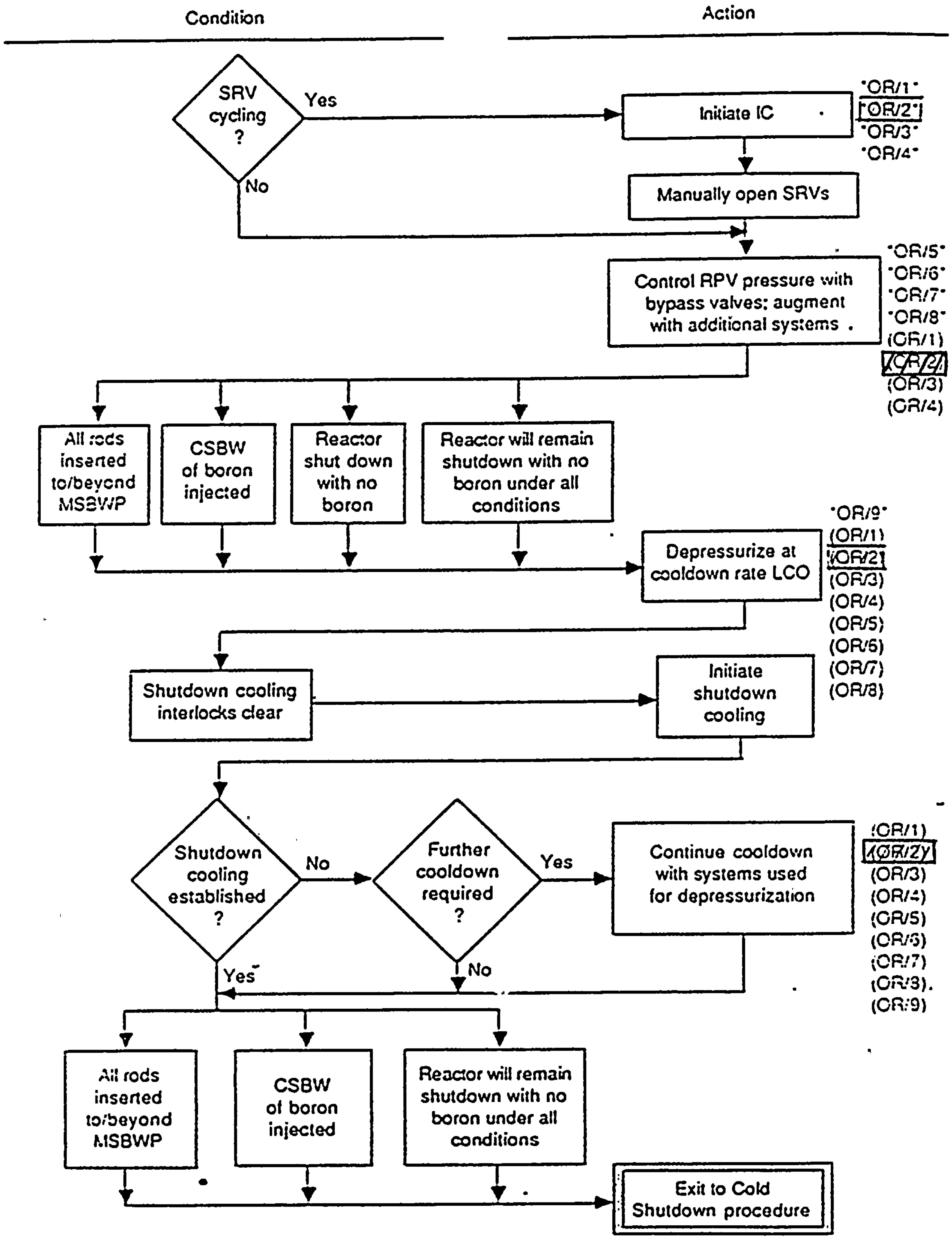


Figure B-6.3: Operator Actions for RPV Pressure Control

DESIGN DEVIATION #6

WNP-2 EOP'S DO NOT DIRECT THE OPERATOR TO DEFEAT THE RPV LOW LEVEL AND HIGH DRYWELL PRESSURE INTERLOCKS TO ALLOW RETURN OF DRYWELL COOLING CAPABILITY.

DEVIATION BASIS

SYMPTOMATICALLY OVERRIDING THE ISOLATION INTERLOCKS TO THE DRYWELL COOLING SYSTEM IS TECHNICALLY WRONG FOR WNP-2 BECAUSE IT MAY CAUSE AN UNNECESSARY RADIATION RELEASE. IN ADDITION, EVEN IF THE INTERLOCKS ARE DEFEATED THE DRYWELL COOLING SYSTEM WILL LIKELY TRIP OR, AT BEST, WILL HAVE A NEGLIGIBLE EFFECT ON LONG TERM HEAT REMOVAL FROM CONTAINMENT.

SAFETY SIGNIFICANCE

A SIGNIFICANT PLANT EVENT (E.G., LOCA) WOULD CAUSE ISOLATION OF THE DRYWELL COOLING SYSTEM. SYMPTOMATICALLY UNISOLATING DRYWELL COOLING IN SUCH AN EVENT COULD CAUSE AN UNNECESSARY RADIATION RELEASE. THE ENERGY REMOVAL CAPABILITY OF THE DRYWELL COOLING SYSTEM IS INSIGNIFICANT WHEN COMPARED TO THE TOTAL ENERGY THAT IS POTENTIALLY AVAILABLE IN CONTAINMENT. IMPACT TO EVENT MITIGATION IS NOT SIGNIFICANT.

DESIGN DEVIATION #7

DIRECTION TO VENT THE PRIMARY CONTAINMENT "BEFORE WETWELL PRESSURE REACHES THE PCPL" HAS BEEN MODIFIED TO ALLOW VENTING WHEN CONTAINMENT PRESSURE IS ABOVE 39 PSIG.

DEVIATION BASIS

VENTING IN THE PRIMARY CONTAINMENT PRESSURE CONTROL PROCEDURE, OCCURS WHEN THE NORMAL MEANS OF DECREASING THE CONTAINMENT PRESSURE HAVE BEEN ATTEMPTED OR ARE IN PROGRESS, THE PSP HAS BEEN EXCEEDED AND EMERGENCY RPV DEPRESSURIZATION HAS COMMENCED.

EPG APPENDIX B TO THE GUIDELINES, PAGE B7-67 STATES THE FOLLOWING WITH RESPECT TO THIS STEP: "NOTE THAT PRIMARY CONTAINMENT VENTING IS PERFORMED ONLY AS NECESSARY TO RESTORE AND THEN MAINTAIN PRESSURE BELOW THE LIMIT."

DEVIATION BASIS (CONTINUED)

THE HIGHEST CONTAINMENT PRESSURE ANALYZED IN ANY OF THE WNP-2 DESIGN BASIS ACCIDENT SCENARIOS IS 39 PSIG. THIS PRESSURE OCCURS APPROXIMATELY AN HOUR INTO THE SCENARIO. IF THE DRYWELL PRESSURE EXCEEDS 39 PSIG, THEN THE ACCIDENT HAS PROGRESSED PAST THE DESIGN BASIS OF THE PLANT AND VENTING THE PLANT BECOMES THE BEST AND SAFEST ACTION TO TAKE.

SAFETY SIGNIFICANCE

THIS DEVIATION IS IN LINE WITH THE CURRENT BWROG STRATEGY, PRESERVES THE WNP-2 DESIGN BASIS, AND REDUCES THE IMPACT TO THE HEALTH AND SAFETY OF THE PUBLIC BY INSURING DIRECT CONTAINMENT VENTING OCCURS FOR ONLY THOSE EVENTS WELL BEYOND THE ANALYZED DESIGN ENVELOP OF WNP-2. THIS APPROACH IS CONSISTENT WITH THE GUIDANCE TAKEN BY WNP-2 FROM THE EPG REV. 4 SER (PAGE 25).

[illegible]

DESIGN BASIS DEVIATION #8

THE BWROG EPG STEPS FOR INITIAL HYDROGEN CONTROL (TO VENT AND PURGE THE DRYWELL TO MAINTAIN COMBUSTIBLE GAS CONCENTRATIONS BELOW MINIMUM DETECTABLE LEVELS) IS NOT GIVEN IN THE WNP-2 PSTG. INSTEAD WNP-2 USES THE INSTALLED HYDROGEN RECOMBINER SYSTEM, TO AVOID POTENTIALLY CONTAMINATED RELEASES FROM PRIMARY CONTAINMENT.

DEVIATION BASIS

AT WNP-2, THE "NORMAL" PROCEDURAL MEANS OF DEALING WITH HYDROGEN IN THE PRIMARY CONTAINMENT ATMOSPHERE IS THROUGH INITIATION OF THE CONTAINMENT ATMOSPHERIC CONTROL (CAC) SYSTEM. INITIATION OF THE CAC SYSTEM ON 0.5% HYDROGEN WILL CONSISTENTLY PROVIDE COMBUSTIBLE GAS CONTROL AS RAPIDLY AS A CONTROLLED PURGE AND VENT SCENARIO WOULD.

THE WNP-2 POSITION IS THAT THE TWO 100% CAPACITY HYDROGEN RECOMBINER SYSTEM WILL BE USED FOR HYDROGEN CONTROL CONSISTENT WITH OUR LICENSING BASIS AS OPPOSED TO CONTAINMENT VENT AND PURGE. THE MAIN IMPACT OF THIS DEVIATION IS TO POTENTIALLY START THE RECOMBINERS EARLIER THAN PRESCRIBED IN THE EPG'S.

SAFETY SIGNIFICANCE

THERE IS NO SAFETY SIGNIFICANCE ASSOCIATED WITH THIS DEVIATION. THE ENDPOINT OF THE WNP-2 EOP'S IS THE SAME AS THE EPG'S. WNP-2 IS DESIGNED AND LICENSED TO CONTROL HYDROGEN CONSISTENT WITH OUR EOP STRATEGY. THIS DEVIATION IS CONSISTENT WITH THE GUIDANCE TAKEN BY WNP-2 FROM THE EPG REV. 4 SER (PAGES 30-31).

DESIGN BASIS DEVIATION #9

RECOMBINER SUCTION FROM DRYWELL WAS NOT EXPLICITLY SPECIFIED IN THE WNP-2 PSTG BECAUSE THIS IS THE NORMAL SYSTEM ALIGNMENT.

DEVIATION BASIS

BECAUSE THE WNP-2 CAC SYSTEM IS NORMALLY ALIGNED WITH SUCTION FROM THE DRYWELL, INCORPORATION OF THOSE WORDS IS NOT REQUIRED. AT WNP-2, CONSIDERABLE EFFORT AND ANALYSES WERE EXPENDED TO SIMPLIFY REQUIRED OPERATOR ACTIONS INCLUDING USE OF THE HYDROGEN RECOMBINER SYSTEM. ONE RESULT OF THIS EFFORT WAS THE CONCLUSION THAT THE SYSTEM COULD RELIABLY CONTROL ANTICIPATED HYDROGEN CONCENTRATIONS IN THE ENTIRE PRIMARY CONTAINMENT WITH A FIXED SYSTEM LINEUP AND CONFIGURATION. WITH THE RECOMBINER SUCTION DRAWN FROM THE DRYWELL, EXHAUSTING TO THE WETWELL AIR SPACE, WITH THE RECYCLE RATE SET AT 55%, THE SYSTEM WILL EFFECTIVELY HANDLE THE FULL SPECTRUM OF ANTICIPATED HYDROGEN CONCENTRATIONS. THE CONTINUOUS ON-LINE HYDROGEN-OXYGEN MONITORING SYSTEM AT WNP-2 PROVIDES RELIABLE INPUT TO THE OPERATORS AT WNP-2 ON DRYWELL AND WETWELL AIRSPACE HYDROGEN AND OXYGEN CONCENTRATIONS.

SAFETY SIGNIFICANCE

THERE IS NO SAFETY SIGNIFICANCE OF THIS DEVIATION AND NO IMPACT ON THE EPG STRATEGY. THE NORMAL ALIGNMENT OF THE RECOMBINER SYSTEM IS WITH SUCTION FROM THE DRYWELL AS SPECIFIED FOR THIS EPG STEP.

DESIGN BASIS DEVIATIONS #10, 11 & 12

RECOMBINER SUCTION FROM WETWELL IS NOT SPECIFIED FOR WNP-2. (INSTEAD HYDROGEN RECOMBINERS ARE OPERATED WITH SUCTION FROM THE DRYWELL WHILE DISCHARGING INTO THE WETWELL.)

HYDROGEN RECOMBINER SYSTEM START PERMISSIVE IS BASED ON "DRYWELL" HYDROGEN/OXYGEN CONCENTRATION RATHER THAN "WETWELL" HYDROGEN/OXYGEN CONCENTRATION.

DIRECTION TO OPERATE THE DRYWELL HYDROGEN MIXING SYSTEM IS GIVEN.

DEVIATION BASIS

THE ABILITY OF THE WNP-2 HYDROGEN RECOMBINER SYSTEMS TO EFFECTIVELY TREAT THE PRIMARY CONTAINMENT AS A "SINGLE" VOLUME HAS BEEN DEMONSTRATED BY TESTS AND PARAMETRIC CALCULATIONS, WHICH SHOW THE EFFECTS OF PROCESSING HYDROGEN CONCENTRATIONS ANTICIPATED UNDER VARIOUS HYDROGEN GENERATION ASSUMPTIONS ENVELOPING THE POSSIBILITIES OF BOTH DRYWELL AND WETWELL AIR SPACE HYDROGEN BUILDUPS. SUCTION FROM THE DRYWELL WITH DISCHARGE TO THE WETWELL WILL CONTROL ACCUMULATION OF COMBUSTIBLE MIXTURES IN EITHER THE DRYWELL OR WETWELL AIRSPACE, EVEN IF THE WETWELL AIRSPACE IS THE SOURCE OF HYDROGEN CONCENTRATIONS.

[illegible]

DEVIATION BASIS (CONTINUED)

THE WETWELL TO DRYWELL VACUUM BREAKERS WILL FUNCTION TO EFFECTIVELY VENT ANY HYDROGEN FROM THE WETWELL TO THE DRYWELL WITH THE CAC OPERATING IN ITS NORMAL ALIGNMENT. BECAUSE THE DRYWELL IS THE SOURCE FOR THE HYDROGEN RECOMBINER SUCTION IT IS APPROPRIATE THAT THE START PERMISSIVE BE BASED ON THE HYDROGEN CONCENTRATIONS IN THE VOLUME BEING PROCESSED.

SAFETY SIGNIFICANCE

BECAUSE THE ACTUAL ENDPOINT OF THESE MODIFIED STEPS ARE THE SAME AS THOSE ANTICIPATED IN THE EPG STRATEGY, AND BECAUSE SUBSEQUENT EPG STRATEGY STEPS ARE FOLLOWED, THERE IS NO SAFETY SIGNIFICANCE TO THIS DEVIATION.

DESIGN DEVIATION #13

DEFEATING ISOLATION INTERLOCKS TO ALLOW RESTART OF THE REACTOR BUILDING HVAC IS NOT SPECIFIED IN THE OVERRIDE WHICH PRECEDES SC/T. WNP-2 EOP'S DO NOT DIRECT THE OPERATOR TO BYPASS THE RPV LEVEL AND HIGH DRYWELL PRESSURE INTERLOCKS TO RETURN THE REACTOR BUILDING HVAC TO OPERATION.

DEVIATION BASIS

PLANT DESIGN PROVIDES FOR A SECONDARY CONTAINMENT ISOLATION ON LOW RPV WATER LEVEL, HIGH DRYWELL PRESSURE AND HIGH SECONDARY CONTAINMENT RADIATION LEVEL. TO DIRECT THE OPERATOR TO BYPASS THE LOW RPV WATER LEVEL AND HIGH DRYWELL PRESSURE INTERLOCK WHENEVER ISOLATION IS CAUSED BY THESE SIGNALS IS CONTRARY TO DEFENSE IN DEPTH CONCEPTS. THE INTENT OF THIS OVERRIDE IN THE BWROG EPG'S IS TO REESTABLISH THE NORMAL MEANS OF COOLING SECONDARY CONTAINMENT TO CONTROL CRITICAL EQUIPMENT AREA TEMPERATURES (OEI DOCUMENT 8390-4 PAGE B-8-25). AT WNP-2 THE TEMPERATURE OF CRITICAL AREAS IN SECONDARY CONTAINMENT IS ADEQUATELY CONTROLLED BY SEPARATE SAFETY GRADE ROOM COOLERS WHICH ARE INITIATED UPON ISOLATION OF SECONDARY CONTAINMENT. UPON INITIATION, THE AREAS COOLED BY THESE EMERGENCY COOLERS ARE AUTOMATICALLY ISOLATED FROM THE NORMAL REACTOR BUILDING HVAC. CONSEQUENTLY, REESTABLISHING THE NORMAL REACTOR BUILDING HVAC WILL NOT PROVIDE OR ENHANCE COOLING IN THESE AREAS.

SAFETY SIGNIFICANCE

IMPLEMENTATION OF THIS DEVIATION WILL NOT SIGNIFICANTLY IMPACT SAFE PLANT OPERATION. THE PRIMARY REASON IS THAT THE SAFETY GRADE AREA AND ROOM COOLERS WILL MAINTAIN THE SECONDARY ENVIRONMENT WITHIN ACCEPTABLE LIMITS. REMOVAL OF THE INTER-LOCKS WILL NOT PROVIDE ANY ADDITIONAL COOLING TO WNP-2 CRITICAL AREAS.

DESIGN DEVIATION #15

PRIOR TO TAKING THE STEP TO FLOOD CONTAINMENT (NON-ATWS ONLY) WNP-2 HAS INSERTED THE PLANT SPECIFIC ANALYZED CONDITION FOR ADEQUATE CORE COOLING (2/3 CORE SUBMERGENCE WITH CORE SPRAY(S) INJECTING AT 6000 GPM).

DEVIATION BASIS

DEVIATION PROVIDES THE OPERATORS THE MAXIMUM AMOUNT OF TIME POSSIBLE BEFORE GOING TO THE STEP OF "LAST RESORT", CONTAINMENT FLOODING.

SAFETY SIGNIFICANCE

WNP-2 VIEWS THIS DEVIATION AS AN EXTENSION OF GENERIC STRATEGY. IT ALLOWS CONTAINMENT FLOODING TO BE DELAYED THEREBY EITHER ELIMINATING OR REDUCING RADIOLOGICAL RELEASES ASSOCIATED WITH FLOODING. OVERALL BWROG STRATEGY OF ADEQUATE CORE COOLING IS MAINTAINED.

DESIGN DEVIATION #16 - DEVIATION WILL BE WITHDRAWN AS PART OF OUR PHASE II REVISION.

HPCS ALLOWED AS RPV INJECTION SOURCE TO FLOOD OR REFILL RPV DURING ATWS, PROVIDED BORON IS BEING INJECTED VIA SLC.

DEVIATION BASIS

WNP-2 IS DESIGNED AND ANALYZED TO USE HPCS SPRAY HEADER AS METHOD FOR INTRODUCING SODIUM PENTABORATE INTO RPV. OPERATION OF HPCS DECREASES TRANSPORT TIME FOR BORON TO REACH CORE AND IS DESIRABLE FOR EVENTS WHEN THE PLANT IS NOT SHUTDOWN, I.E., ATWS.

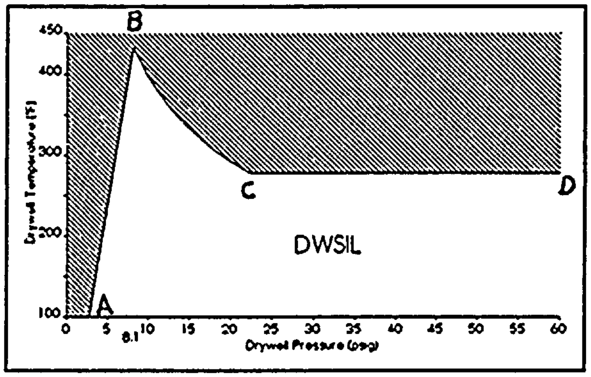
SAFETY SIGNIFICANCE

HPCS INJECTION WITH SLC IS BORATED TO GREATER THAN SHUTDOWN BORON CONCENTRATIONS, ANALYSES INDICATE POWER EXCURSIONS WITH BORATED HPCS INJECTION ARE MINIMAL.

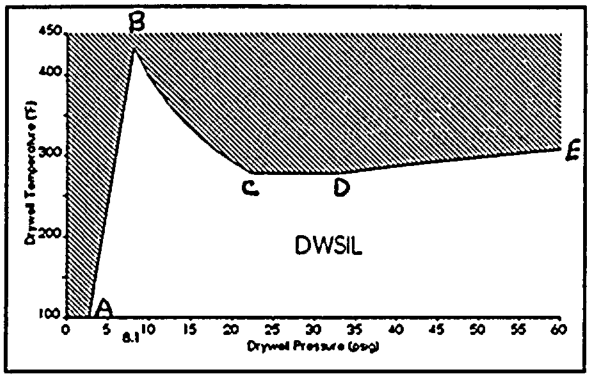
DUE TO INABILITY TO DEMONSTRATE BY ANALYSES THAT THIS DEVIATION IS OPTIMUM STRATEGY FOR ALL SLC INJECTION EVENTS, WNP-2 WILL WITHDRAW THIS DEVIATION IN PHASE II REVISION.

DESIGN DEVIATION #17

DWSIL CALCULATION SLIGHTLY MODIFIED TO ALLOW USE OF DRYWELL SPRAYS IN SATURATED ATMOSPHERE CONDITIONS AT HIGH PRESSURES.



Generic DWSIL

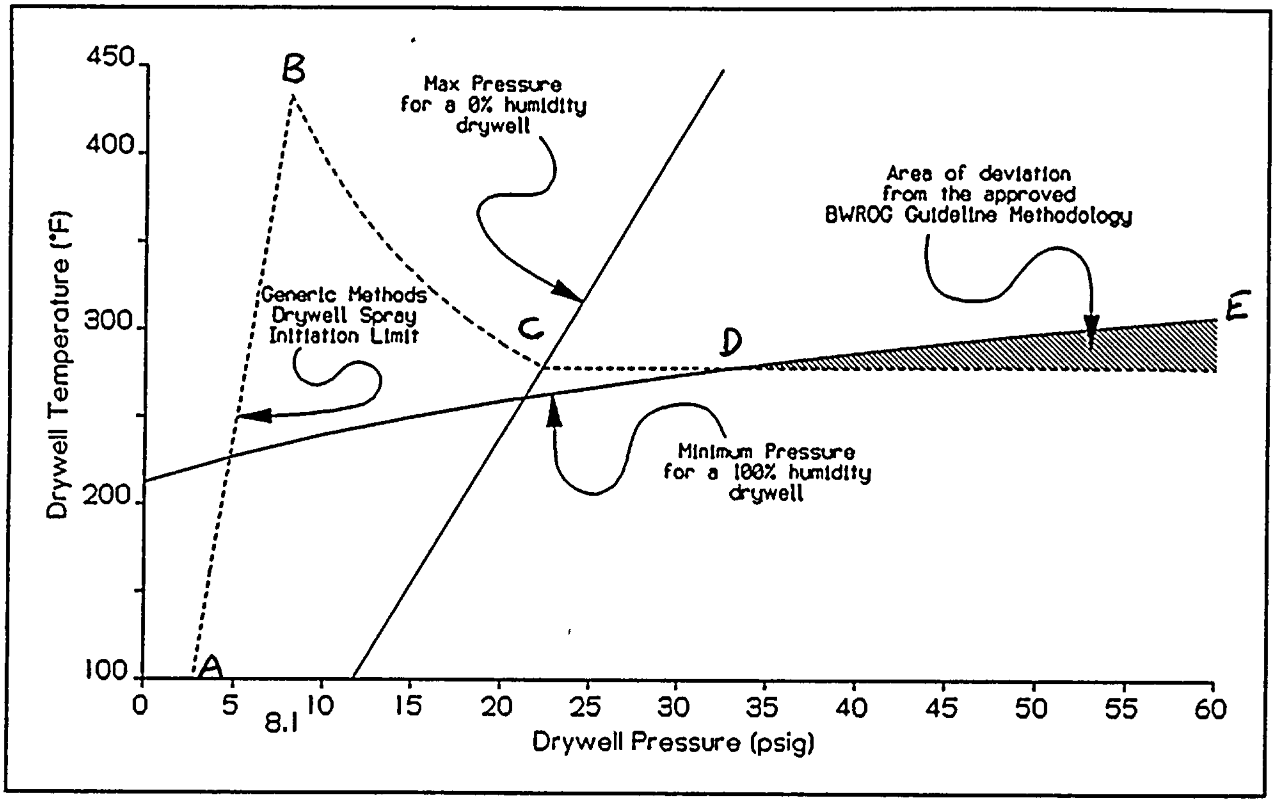


WNP-2 DWSIL

DEVIATION BASIS

BWROG EPG APP. C DWSIL WORKSHEET (WS-3) RESULTED IN A CURVE FOR WNP-2 WHICH PREVENTS USE OF DW SPRAYS WHEN WNP-2 LICENSING BASIS ASSUMES THEY WILL BE USED.

WNP-2 ADDED REGION IS OUTSIDE CONFINES OF ASSUMPTIONS IN EPG METHODOLOGY. NO CONFLICT WITH EPG'S IN ADDING THIS REGION TO DWSIL CURVE. OPERATING MARGIN GAINED AND ORIGINAL FUNCTION OF DW SPRAY HEADER MAINTAINED.



SAFETY SIGNIFICANCE

- USEABLE REGION OF DWSIL CURVE FOR DW SPRAYS EXPANDED WITHOUT VIOLATING BASIS ASSUMPTIONS FOR THE CURVE,

- ALLOWS MITIGATION OF SOME LOCA HEATING SCENARIOS THAT WOULD NOT OTHERWISE HAVE BEEN POSSIBLE.

- PLANT SPECIFIC ANALYSIS SUPPORTS SPRAYING IN THIS ADDED REGION.

- ACCOMMODATE WNP-2 LICENSING ANALYSIS.

STRATEGY DEVIATIONS - 6 TOTAL

APPROACH

- WNP-2 BELIEVES EPG STRATEGIES ARE APPROPRIATE.
- NO EPG STRATEGY WAS COMPLETELY REPLACED.
- EPG STRATEGIES SUPPLEMENTED TO ALLOW SIMPLIFICATION BY TAKING ADVANTAGE OF WNP-2 DESIGN FEATURES OR TO ENHANCE HUMAN FACTORS ASPECTS OF THE OPERATOR IMPLEMENTATION.

STRATEGY DEVIATION #1

IN THE EVENT THE NONSEISMIC CST'S, WHICH ARE DESIGNATED AS PREFERRED RCIC SUCTION SOURCE ARE LOST, CONTINUED RCIC OPERATION WITH SUCTION FROM SUPPRESSION POOL WOULD BE ALLOWED.

DEVIATION BASIS

DUE TO NONSEISMIC DESIGN OF CST'S AND ASSOCIATED SUCTION PIPING, WNP-2'S CST'S MAY NOT BE AVAILABLE FOR ALL CONDITIONS ASSUMED OR INTENDED BY GENERIC GUIDANCE. ALLOWING RCIC TO TAKE SUCTION FROM SP UPON LOSS OF CST SUCTION INCREASES AVAILABILITY OF RCIC. POSITION IS IN LINE WITH APPROVED BWROG EPG ISSUE 8905.

<u>SAFETY SIGNIFICANCE</u>

- TAKING ADVANTAGE OF SP SUCTION WHEN CST'S LOST, INCREASES RCIC AVAILABILITY FOR RPV MAKEUP.

- ALL RCIC CONTROL LOGIC AND SYSTEM COMPONENTS ARE POWERED FROM CLASS 1E BATTERIES ALLOWING FLEXIBILITY TO REALIGN SUCTION UNDER EVEN AN SBO.

- DURING SBO, SP SUCTION MAY IMPROVE RCIC RELIABILITY BY PREVENTING INCREASED RCIC TURBINE BACK PRESSURE FROM SP LEVEL INCREASES.

STRATEGY DEVIATION #3

WNP-2 DESIGNATES PURGING OF THE PRIMARY CONTAINMENT WITH NITROGEN INSTEAD OF WITH NITROGEN OR AIR WHEN A FLAMMABLE MIXTURE OF H_2 AND O_2 EXISTS IN PRIMARY CONTAINMENT.

DEVIATION BASIS

IN THE EVENT THE TWO 100% CAC SYSTEMS ARE UNAVAILABLE OR UNABLE TO CONTROL THE COMBUSTIBLE GAS BUILDUP, WNP-2 WILL UTILIZE ITS INSTALLED NITROGEN INERTING SYSTEM TO PURGE CONTAINMENT. THIS SYSTEM IS DESIGNED TO DELIVER NITROGEN AT PRESSURES UP TO AND BEYOND THE ULTIMATE CONTAINMENT RUPTURE POINT (~130 PSI) EVEN UNDER SBO CONDITIONS. EVEN IF THE CONTAINMENT WERE DEINERTED, ADDING NITROGEN WOULD BE PREFERABLE TO ADDING AIR, AS IT WOULD BOTH DILUTE THE HYDROGEN AND REDUCE THE SEVERITY OF ANY DEFLAGRATION THAT COULD OCCUR.

AT WNP-2 THE REACTOR BUILDING HVAC FANS THAT COULD BE USED FOR AN AIR PURGE HAVE A MAXIMUM DISCHARGE PRESSURE OF 0.5 PSI. BECAUSE ACCIDENTS AT WNP-2 THAT COULD GENERATE SIGNIFICANT QUANTITIES OF COMBUSTIBLE GASES WILL ALSO PRESSURIZE PRIMARY CONTAINMENT ABOVE 0.5 PSI, USE OF THE AIR PURGE FANS IS BELIEVED TO BE OF LIMITED USE.

DEVIATION BASIS (CONTINUED)

THE WNP-2 AIR PURGE FANS ARE ALSO ELECTRICALLY SHED IN THE EVENT OF AN ACCIDENT AND CANNOT BE RESTORED UNLESS OFFSITE POWER IS AVAILABLE. THE NITROGEN PURGE SYSTEM REMAINS AVAILABLE POST-ACCIDENT. DESIGNATION AND PREFERENTIAL USE OF THE NITROGEN PURGE IN THE WNP-2 EOP'S ADEQUATELY COVER THE ANTICIPATED SPAN OF DESIGN BASIS AND SEVERE ACCIDENTS. SHOULD THE NITROGEN SOURCES NOT BE AVAILABLE, OR SHOULD OTHER EMERGENCY CONDITIONS WARRANT DIFFERENT ACTIONS, THE DECISION TO PURGE THE CONTAINMENT WITH AIR COULD BE MADE BY THE IN-PLACE EMERGENCY ORGANIZATION AT THE TIME.

SAFETY SIGNIFICANCE

NO ADVERSE SAFETY SIGNIFICANCE IS INVOLVED IN IMPLEMENTING THIS DEVIATION BECAUSE THE OBJECTIVE OF PURGING AND VENTING IS MET USING A NITROGEN SOURCE PREFERENTIALLY. THIS ACTION IS NOT INCONSISTENT WITH THE EPG STRATEGY AND IF SUCCESSFUL RESULTS IN DILUTION OF HYDROGEN IN CONTAINMENT WITH NITROGEN TO DIMINISH THE EFFECTS OF ANY DEFLAGRATION. AT WNP-2, THE NITROGEN PURGE, UNLIKE THE AIR PURGE OPTION, IS AVAILABLE FOR CONTAINMENT PRESSURES GREATER THAN 0.5 PSI AND FOR LOSS OF OFFSITE POWER EVENTS.

STRATEGY DEVIATION #4

WHEN ISOLATING PRIMARY SYSTEMS THAT ARE DISCHARGING INTO A SECONDARY CONTAINMENT AREA OR OUTSIDE THE PLANT, THE BWROG REV. 4 GUIDELINES EXCLUDE ALL SYSTEMS REQUIRED TO SHUT DOWN THE REACTOR, ASSURE ADEQUATE CORE COOLING, OR SUPPRESS A FIRE. IN KEEPING WITH THE LATEST BWROG EPG COMMITTEE STANCE, WNP-2 WILL ALSO EXCLUDE SYSTEMS REQUIRED TO VENT THE CONTAINMENT IN THIS STEP.

DEVIATION BASIS

BY REQUIRING THE ISOLATION OF ONLY THOSE SYSTEMS NOT REQUIRED TO ASSURE ADEQUATE CORE COOLING, REGARDLESS OF THEIR CURRENT EFFECT ON THE SECONDARY CONTAINMENT, THE BWROG EPG HAS SET THE PROTECTION OF THE CORE AHEAD OF SECONDARY CONTAINMENT.

PAGE B-7-71 OF APPENDIX B OF THE BWROG GUIDELINES STATES "WHEN A DECISION BETWEEN THE POSSIBLE LOSS OF ADEQUATE CORE COOLING AND A LOSS OF PRIMARY CONTAINMENT INTEGRITY MUST BE MADE, THE EPG'S PREFERENTIALLY CHOOSE TO MAINTAIN PRIMARY CONTAINMENT INTEGRITY..." THEREFORE, IT DIRECTLY FOLLOWS THAT THE PRIMARY CONTAINMENT TAKES PRECEDENCE OVER THE SECONDARY CONTAINMENT.

[illegible]

SAFETY SIGNIFICANCE

THIS DEVIATION CORRECTLY PRIORITIZES PRIMARY CONTAINMENT OVER SECONDARY CONTAINMENT CONSISTENT WITH EPG INTENT. WNP-2 POSITION IS IN COMPLIANCE WITH LATEST BWROG RESOLUTION (EPG #8902).

STRATEGY DEVIATION #5

THE SAME GUIDANCE GIVEN IN THE BWROG EPG FOR RESTART OF THE TURBINE BUILDING HVAC SYSTEM IS ALSO GIVEN IN THE WNP-2 PSTG FOR RESTART OF THE RADWASTE BUILDING HVAC SYSTEM.

DEVIATION BASIS

THE EPG'S DO NOT DIRECT RESTART OF THE RADWASTE BUILDING HVAC SYSTEM. HOWEVER, AT WNP-2, THERE ARE POTENTIAL RELEASE PATHS THAT TERMINATE IN THE RADWASTE BUILDING (SUCH AS THE RWCU, EDR AND FDR SYSTEMS), AND PERSONNEL ACCESS TO THIS BUILDING (WHICH HOUSES THE MAIN AND RADWASTE CONTROL ROOMS) IS ALSO DESIRABLE.

EPG STRATEGY DID NOT ADDRESS ANY BUILDING OTHER THAN THE TURBINE BUILDING BECAUSE IT WAS CERTAIN ONLY THAT ALL BWR'S HAVE A TURBINE BUILDING. RESOLUTION OF EPG ISSUE #53 CONTAINS THE FOLLOWING QUOTE CONCERNING "OTHER" BUILDING HVAC RESTART: "IF A UTILITY BELIEVES VENTILATION SYSTEMS IN OTHER AREAS FALL INTO THE SAME CATEGORY AS TURBINE BUILDING HVAC, THEY TOO COULD BE INCLUDED IN THIS STEP."

SAFETY SIGNIFICANCE

THERE IS NO ADVERSE SAFETY SIGNIFICANCE ASSOCIATED WITH IMPLEMENTING THIS DEVIATION. THIS IS MERELY AN ADDED STEP THAT WILL NOT ADVERSELY AFFECT ANY OTHER MITIGATING ACTION OR EQUIPMENT, AND INSURES THAT ANY RELEASE FROM THE RADWASTE BUILDING IS ELEVATED AND MONITORED.

[illegible]

STRATEGY DEVIATION #6

WITH NO INJECTION SYSTEMS AVAILABLE, WNP-2 WILL NOT EXECUTE EPG LOW PRESSURE OVERRIDE WHICH RESULTS IN EARLY TERMINATION OF STEAM COOLING. WNP-2 WILL CONTINUE IN STEAM COOLING, IRRESPECTIVE OF RPV PRESSURE UNTIL EITHER ANY INJECTION SYSTEM BECOMES AVAILABLE OR RPV LEVEL DROPS TO MIN ZERO INJECTION WATER LEVEL.

DEVIATION BASIS

LITERAL EXECUTION OF EPG GUIDANCE WOULD REQUIRE OPERATOR TO EXIT STEAM COOLING BASED ON HIGHEST DISCHARGE PRESSURE OF PLANT'S ALTERNATE INJECTION SYSTEM EVEN IF NO INJECTION SYSTEMS AVAILABLE. EPG APPROACH MAKES ARBITRARY TIE BETWEEN DISCHARGE HEAD OF ALTERNATE INJECTION SYSTEMS AND POINT WHERE STEAM COOLING BECOMES INEFFECTIVE. EARLY TERMINATION PRECLUDES PLANT FROM TAKING ADVANTAGE OF STEAM COOLING WHEN IT IS STILL EFFECTIVE & REDUCES THE TIME AVAILABLE TO OPERATORS TO RECOVER INJECTION SYSTEMS.

SAFETY SIGNIFICANCE

NUSCO ANALYSIS DEMONSTRATES STEAM COOLING REMAINS EFFECTIVE DOWN TO VERY LOW RPV PRESSURES. EXTENDING STEAM COOLING TO LOW RPV PRESSURES COUPLED WITH EMERG. DEPRESS. SHOULD AN RPV INJECTION SOURCE BE RECOVERED, MAINTAINS OVERALL INTENT OF EPG STRATEGY. PROVIDES OPERATOR MAX TIME TO RECOVER INJECTION SYSTEM.

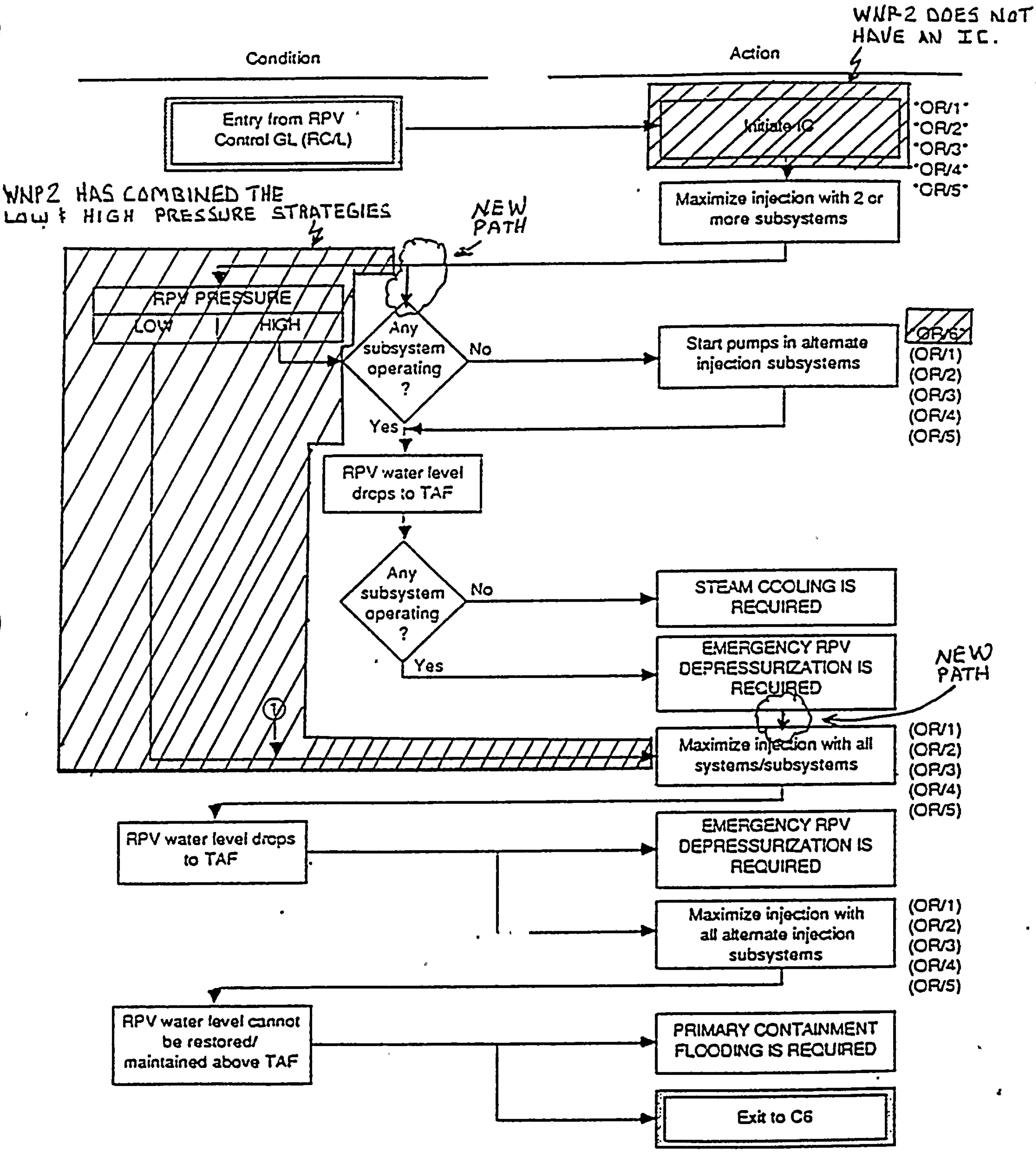


Figure B-10.1: Operator Actions for Alternate Level Control

STRATEGY DEVIATION #8

DURING AN ATWS, WNP-2 HAS INCLUDED A HOLDPOINT WHICH REQUIRES OPERATOR TO CONFIRM THAT ONE OR MORE SLC PUMPS ARE NOT RUNNING OR THAT REACTOR POWER IS NOT DECREASING PRIOR TO INITIATING ACTION TO INTENTIONALLY REDUCE RPV WATER LEVEL.

DEVIATION BASIS

BEST ESTIMATE ATWS ANALYSES INDICATE REACTOR POWER IS RAPIDLY REDUCED WHEN SODIUM PENTABORATE IS INJECTED VIA HPCS SPRAY HEADER INTO CORE. CONTROL OF REACTOR POWER BY THIS METHOD IS PREFERRED TO REDUCTION IN RPV WATER LEVEL WHICH REQUIRES OPERATOR INTERVENTION TO OVERRIDE AUTOMATIC SYSTEM CONTROLS AND THROTTLE OR TERMINATE INJECTION OF HIGH PRESSURE MAKEUP SYSTEMS.

SAFETY SIGNIFICANCE

ADDING SLC PUMP OPERATION AND REACTOR POWER DECREASING AS CONDITIONS TO REDUCING RPV WATER LEVEL DOES NOT PRECLUDE EPG ACTION FROM OCCURRING IF NECESSARY. IF EITHER ADDED CONDITION FAILS, ACTION TO REDUCE LEVEL IS INITIATED. PARAMETRIC ANALYSES SHOW SIGNIFICANT CONTAINMENT DESIGN MARGIN RETAINED EVEN WITH DELAYED OPERATOR ACTION.

[illegible]
[illegible]