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MAY 2 2 2001

U.S. Nuclear Regulatory Commission
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Mail Station OP1-17
Washington, DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION SUPPLEMENTAL INFORMATION APPLICABLE TO PROPOSED AMENDMENT NO. 235 TO LICENSE NPF-14 AND PROPOSED AMENDMENT NO. 200 TO LICENSE NPF-22: POWER UPRATE PLA-5300

Docket No. 50-387 and 50-388

Reference:
1) *PLA-5276, R. G. Byram To USNRC, Revised Submittal of Proposed Amendment No. 235 to License NPF-14 and Proposed Amendment No. 200 to NPF-22: Power Uprate dated 02/08/2001*
2) *NRC RAI, R. G. Schaaf to R. G. Byram, "Request for Additional Information Regarding 1.4 – Percent Power Uprate (TAC NOS. MB0444 and MB0445) dated 04/30/2001*

The purpose of this letter is to respond to your Request for Additional Information (RAI) [Reference 1] and to describe changes to the power ascension test regime delineated in reference 1.

The RAI questions and our responses are contained in Attachment 1.

The power ascension testing changes are described in Attachment 2.

The No Significant Hazards Considerations and Environmental Assessment provided in Reference 1 are not affected by the information provided herein.

PPL Susquehanna, LLC requests approval of the proposed Amendment prior to June 1, 2001.

Contained herein are the following two PPL commitments:

A001

PLA-5300-1

Prior to implementation of the Power Uprate on Unit 1 in the Spring 2002, PPL commits to implement modifications on the Unit 1 SLC system so that the SLC ATWS analysis remains valid for Unit 1.

PLA-5300-2

PPL commits to revise the Unit 1 and Unit 2 P/T curves. The revised curves will be submitted by August 30, 2001. These curves will contain a note that will identify that they are valid until May 2006 and May 2005 for Unit 1 and Unit 2 respectively.

If you have any questions, please contact Mr. M. H. Crowthers at (610) 774-7766.

Sincerely,

G. T. Jones

Attachment

copy: NRC Region I
Mr. S. Hansell, NRC Sr. Resident Inspector
Mr. R. G. Schaaf, NRC Project Manager
Mr. D. J. Allard, PA DEP

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC: Docket No. 50-387

SUPPLEMENTAL INFORMATION APPLICABLE TO
PROPOSED AMENDMENT NO. 235 TO LICENSE NPF-14:
POWER UPRATE
UNIT NO. 1

Licensee, PPL Susquehanna, LLC, hereby files supplemental information in support of a revision to its Facility Operating License No. NPF-14 dated July 17, 1982.

This amendment involves a revision to the Susquehanna SES Unit 1 Technical Specifications.

PPL Susquehanna, LLC
By:

GT Jones

G. T. Jones
Vice-President - Nuclear Engineering & Support

Sworn to and subscribed before me
this 22nd day of May, 2001.

Nancy J. Lannen

Notary Public

Notarial Seal
Nancy J. Lannen, Notary Public
Allentown, Lehigh County
My Commission Expires June 14, 2004

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC : Docket No. 50-388

SUPPLEMENTAL INFORMATION APPLICABLE TO PROPOSED AMENDMENT NO. 200 TO LICENSE NPF-22: POWER UPRATE UNIT NO. 2

Licensee, PPL Susquehanna, LLC, hereby files supplemental information in support of a revision to its Facility Operating License No. NPF-22 dated March 23, 1984.

This amendment involves a revision to the Susquehanna SES Unit 2 Technical Specifications.

PPL Susquehanna, LLC
By:

G. T. Jones
G. T. Jones
Vice-President - Nuclear Engineering & Support

Sworn to and subscribed before me
this 22nd day of May, 2001.

Nancy J. Lannen
Notary Public

Notarial Seal
Nancy J. Lannen, Notary Public
Allentown, Lehigh County
My Commission Expires June 14, 2004

Attachment 1 to PLA-5300
Supplemental Information

Supplemental Information

Question 1

You stated in your application that the approach, scope and detail of your power uprate evaluation are based on the General Electric (GE) generic boiling-water reactor power uprate guidelines presented in Licensing Topical Reports LTR1[1] and LTR2[2], and the specific design features of the SSES units. You also stated that the cores for both units in the upcoming cycles would consist exclusively of Siemens Power Corporation (SPC) Atrium-10™ fuel bundles. Please explain the impact the 1.4-percent power increase and the SPC Atrium 10™ core have on the minimum critical power ratio safety limit values for both units. Please provide the cycle-specific reload safety analyses supporting operation at the uprated conditions (e.g., Final Safety Analysis Report, Appendix 15D). Also, identify any operating flexibility options for which SSES Units 1 and 2 may be licensed and discuss the impact, if any, the power uprate may have on operation under these conditions.

Response:

Methodology

The cycle specific MCPR Safety Limit analysis is performed by Framatome-ANP (FRA-ANP -- formerly Siemens Power Corporation) using the NRC approved methodologies described in References 1 and 2. These references are listed in Section 5.6.5 of the Technical Specifications for both Unit 1 and Unit 2.

The analysis consists of a statistical (Monte Carlo) combination of thermal margin related uncertainties. These uncertainties are feedwater flow, feedwater temperature, core pressure, core flow, assembly flow rate, radial bundle power, local power, axial power, and the critical power correlation. Additionally, power distributions throughout the cycle are calculated by PPL based on the core design for the cycle of interest and transmitted to FRA-ANP for use in the analysis. This is illustrated in Figure 5.1 of Reference 1. A value of the Safety Limit MCPR is also input to the calculation. The calculation output is the number of fuel pins expected to be in boiling transition. If the calculated number of pins in boiling transition is > 0.1% of the total number of fuel pins in the core, then the assumed Safety Limit is increased by 0.01, and the calculation is repeated. The safety limit is determined once the number of pins in boiling transition is predicted to be ≤ 0.1% of the total number of fuel pins.

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1 GE Licensing Topical Report NEDC-31897P-A, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," May 1992.

2 GE Licensing Topical Report NEDC-31984P, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," July 1991.

The calculation of MCPR Safety Limits (two-loop and single-loop) for each unit and cycle preserves the margin of safety described in Section B2.1.1 of the Technical Specification Bases. The safety limit is added to the cycle specific ΔCPRs (calculated using NRC approved methodology) to produce the MCPR Operating Limits contained in the Core Operating Limits Reports.

Factors Affecting Changes to the MCPR Safety Limit

Three factors could potentially affect the difference in the MCPR Safety Limits for the proposed power uprates (3489 MWt) for Unit 2 Cycle 11 and Unit 1 Cycle 13: the transition from a mixed core (w/ FRA-ANP 9x9-2 fuel) to an all ATRIUM™-10 core, the increase in rated power, and the core design. This discussion is provided below.

In the cycle immediately preceding the power uprate cycle, the 9x9-2 assemblies were high exposure / low power assemblies that do not contribute any calculated pins in boiling transition. Thus, the transition from a mixed core (containing 9x9-2 and ATRIUM™-10 fuel) to the U2C11 and U1C13 (all ATRIUM™-10) cores does not affect the calculated MCPR Safety Limits.

For a given core configuration, an increase in core power flattens the core radial power distribution due to void feedback, and a flatter distribution (more bundles having peaking factors close to the maximum peaking factor) will increase the number of pins calculated to be in boiling transition. However, since the increase in rated power is only 1.4% power, the impact on the core power distribution is very small. Thus, the small increase in rated power is at most a very minor contributor to the increase in the calculated MCPR Safety Limit.

Past reload analyses in which no change in rated core power occurred have resulted in changes to the MCPR Safety Limit (~ .01 to .02). When designing a core with a power uprate, there is a tendency to design the core with lower bundle radial peaking factors in order to increase MCPR operational margin. As stated above, this tends to result in more pins calculated to be in boiling transition for a given value of the Safety Limit – thus, a higher MCPR Safety Limit might be required as a result of a power uprate core design.

Thus, it is likely that the change in MCPR Safety Limit is mainly due to the cycle specific core design.

Plant Response for Power Uprate

The U2C11 cycle specific transient analyses (showing the transient response of the reactor) will be incorporated into the SSES FSAR upon NRC approval of the proposed uprate. The U1C13 FSAR changes are expected to be similar but will not be developed or effective until implementation in Spring 2002. Included herein are mark-ups of the current FSAR reflecting the change from U2C10 (3441 MWt rated power) to U2C11 (3489 MWt rated power). The changes in plant response are relatively minor.

Except for the ability to operate at a slightly higher power, Single Loop Operation is virtually unaffected by the proposed increase in rated core power. The single-loop pump seizure event was calculated and is shown in the FSAR mark-ups.

Effect of Power Uprate on Plant Flexibility Options

SSES is licensed to operate with extended load line limit analysis (ELLLA), with Increased Core Flow (maximum core flow of 108 Mlb/hr), and with Single Loop Operation. Final Feedwater Temperature Reduction is not presently a licensed option at SSES. With the exception of the power uprate, no additions to the SSES options are proposed.

The U2C11 licensing analyses were performed for the proposed power uprate based on the ranges of power and flow allowed by the Power/Flow Map, which reflect the ELLLA and Increased Core Flow ranges. Thus, the analysis results (see included FSAR mark-ups) and COLR operating limits support continued operation with both ELLLA and Increased Core Flow.

Except for the ability to operate at a slightly higher power, Single Loop Operation is virtually unaffected by the proposed increase in rated core power. Using this higher power level, the single-loop pump seizure event was explicitly analyzed for U2C11 and the results are shown in the FSAR mark-ups included herein.

References

1. *ANF-524(P)(A), Revision 2 and ANF-524(P)(A) Supplement 1, Revision 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors".*

2. *EMF-1997(P)(A), Revision 0, "ANFB-10 Critical Power Correlation" and EMF-1997(P)(A) Supplement 1, Revision 0, "ANFB-10 Critical Power Correlation: High Local Peaking Results."*

Question 2

In your previous power uprate submittal (Reference 1.6[3] of NE-2000-00-1P), you stated that "SLCS [standby liquid control system] shutdown capability is evaluated for each fuel reload ... A small increase in the SRV [safety/relief valve] setpoint has no effect on the rated injection flow to the reactor, and the resulting increased system operating pressure has not reduced the SLC pump relief valve pressure margin below the recommended

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[3] PPL Susquehanna, LLC, Licensing Topical Report NE-092-001 Rev. 0, "Susquehanna Steam Electric Station Units 1 and 2, Power Uprate With Increased Core Flow," June 1992.

levels. Therefore, the capability of the SLCS to provide its backup shutdown function is not affected by the power uprate... A similar evaluation confirmed that the SLC will continue to meet the requirements of 10 CFR 50.62 for ATWS [anticipated transient without scram]." For the currently proposed power uprate, you stated that "an evaluation is performed to assure that the SLCS continues to meet the requirements of 10 CFR 50.62 for ATWS."

- What are, (1) the limiting ATWS transients, (2) the peak steam dome pressure, and (3) the required discharge pressure for the SLC pumps? Submit actual analyses that evaluate the response and the injection capabilities of the SLC and reactor core isolation cooling systems during the limiting ATWS transient at the uprated condition.

Response:

SSES FSAR Chapter 15.8 details the SSES Unit 1 and Unit 2 response to the Anticipated Transients without Scram events. It identifies that seven initiating events are considered, one of which is a loss of normal AC power.

Section 15.8.1.4.1 identifies that the "most severe ATWS events are initiated by a pressurization transient (MSIV closure of turbine trip) or by an equipment failure which leads to a pressurization transient (e.g. pressure regulator failure; loss of condenser vacuum).

A discrepancy was recently discovered affecting the loss of normal AC power transient analysis. This discrepancy affected the SLC pump discharge pressure during the transient. This discrepancy has been addressed via the PPL corrective action program. As a result, modifications have been implemented on Unit 2 during the U2 10RIO in the Spring 2001. These modifications ensure that the SLC system will inject as previously assumed in the FSAR for the loss of normal AC power transient such that the most severe ATWS event is the MSIV closure transient as described in the SSES FSAR Section 15.8. Installation of the modification resolves the discrepancy so that the ATWS analysis shows conformance to the 10 CFR 50.62 ATWS requirements.

Analysis conclusions reflected in the SSES Power Uprate Licensing Topical Report NE-2000-001P Rev. 1 and the SSES FSAR Section 15.8 regarding the ability of SLC to inject as assumed in ATWS analyses remain valid for Unit 2.

Regarding Unit 1, prior to implementation of the Power Uprate on Unit 1 in the Spring 2002, PPL commits to implement modifications on the Unit 1 SLC system so that the SLC ATWS analysis also remains valid for Unit 1.

Question 3

You stated in your submittal that because the uprated power does not entail an increase in the operating pressure used for evaluation, the SRV pressure setpoints do not have to be changed. Please verify that the SRV's can provide the necessary overpressure protection during limiting anticipated operational occurrence transients, ATWS transients, and American Society of Mechanical Engineers (ASME) overpressure transients.

Response

Transient Overpressure (TOP) analysis, performed specifically for Unit 2 Cycle 11 at the uprated conditions, is documented in draft updates to Chapter 5.2 of the FSAR contained herein. This analysis demonstrates that the SRV's provide the necessary overpressure protection with respect to the TOP limits. In addition, Chapter 15 analysis has also been performed specifically for Unit 2 Cycle 11 operation at the uprated conditions, and results meet all criteria set forth in Chapter 15. Therefore, SRV setpoints do not require revision due to operation at a licensed power level of 3489 MWt.

Question 4

Section 3.3.1 of PPL Susquehanna, LLC, Report NE-2000-001P states that "...based on the expected increase and the conservative evaluation... the pressure versus temperature (PT) curves...are unchanged and remain bounding."

- ME-2000-001P and References 1.6 and 3.1[4]of NE-2000-001P do not discuss any sources of conservatism in the evaluations. Please clarify and support the argument that there is sufficient conservatism to justify that the PT curves remain unchanged.

- The fluence values were based on a dosimetry reports[5] by the Southwest Research Institute (SwRI) published in 1986. There have been many changes in cross sections and analytical techniques since that time. Please provide information to support the assertion that the original values are conservative for the proposed application.

- There is no dosimetry referenced for Unit 2, thus, the evaluations for both units are based on a single capsule measurement for Unit 1. Please address the adequacy of only one dosimetry measurement.

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4 GE Report SASR 89-11, "Implementation of Regulatory Guide 1.99, Revision 2 for Susquehanna Steam Electric Station Units 1 and 2," May 1989.

5 SwRI Report 06-8658, "Susquehanna Unit 1 Dosimeter Testing," September 1986.

Response

The original estimate of the 32 EFPY fluence at the maximum location in the vessel wall surface was calculated by GE at the time of SSES construction to be 1.1E18 n/cm^2. This was based on calculated fluxes from a generic BWR vessel calculation and the RTndt values obtained from impact and drop-weight tests performed at that time for the vessel materials. After the end of the first cycle of operation for Unit 1, the dosimeter capsule was withdrawn from the 30 degree location and sent to Southwest Research Institute Laboratories for evaluation. Southwest also performed a specific fluence calculation based on the current calculation methodology and a specific model of the Susquehanna reactor. This model included a box model of the jet pump that resided directly in front of the dosimeter location in the vessel. The computer transport calculations performed for 32 EFPY fluence produced a value of 7.74E17 n/cm^2 or 70% of the original GE value. Southwest also reported that their dosimetry evaluation produced fluences that were 8% lower than the numerical calculation results. P/T curves were produced from this evaluation based on the numerical calculation value rather than the dosimeter fluence for conservatism sake. A specific error evaluation for their computer program output was not provided.

PLA-2852 dated May 8, 1987 (included herein), addresses the missing Unit 2 neutron dosimeter. GE report SASR 89-11, on page 2-3, also discusses briefly why the dosimeter values for Unit 1 could be used for Unit 2. In summary, it says that the two vessel geometry's are essentially identical and the core power shapes are similar. Therefore, the Unit 1 dosimeter values adequately serve as a best estimate for the Unit 2.

As described above, the current curves are not based on current methodologies. Since they are not based on current methodologies, PPL commits to revise the Unit 1 and Unit 2 P/T curves per RG 1.190 by May 2006 and May 2005 for Unit 1 and Unit 2. respectively. In the interim, PPL will submit new P/T curves revised to ASME code case N-640 by the Summer of 2001 with the notation that they will be valid only to May 2006 and May 2005 for Unit 1 and Unit 2 respectively based on the most recent fluence evaluations performed at the last surveillance capsule testing.

The May 1, 2006 and May 1, 2005 time limits for Unit 1 and Unit 2 respectively were chosen to allow time for performance of any reanalysis by PPL pursuant to the new Regulatory Guide, development of subsequent proposed T.S. changes, and time for NRC review\approval of proposed revised curves.

Question 5

The Nuclear Regulatory Commission (NRC) staff's safety evaluation dated March 8, 1999, regarding "Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM [leading edge flow meter] System," Included 4 criteria that licensees need to address when referencing the topical report. Criteria 3 states:

> The licensee should confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative methodology is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

Please provide a copy of your comparison of the uncertainty for the LEFM system to the current feedwater instrumentation for NRC staff review.

Response

The calculation included herein, EC-031-1010 revision 0, provides the methodology used and the comparison results.

Question 6

Nuclear power plants are licensed to operate at a specified power, which, at operating power levels, is indicated in the control room by neutron flux instrumentation that has been calibrated to correspond to core thermal power. Core thermal power is determined by a calculation of the energy balance of the plant nuclear steam supply system. The accuracy of this calculation depends primarily upon the accuracy of feedwater flow, feedwater enthalpy, and main steam enthalpy measurements, which are not safety grade and are not included in the plant technical specifications.

The uncertainty of calculating values of core thermal power determines the probability of exceeding the power levels assumed in the design-basis transient and accident analyses. In this regard, to allow for uncertainties in determining thermal power (e.g., instrument measurement uncertainties), Appendix K to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50) requires loss-of-coolant accident and emergency core cooling system (ECCS) analyses to assume that the reactor had operated continuously at a power level at least 102 percent of the licensed thermal power. The 2-percent power margin uncertainty value was intended to address uncertainties related to heat sources in addition to instrument measurement uncertainties. Later, the NRC concluded that, at the time of the original ECCS rulemaking, the 2-percent power margin requirement appeared to be based solely on considerations associated with power measurement uncertainty.

Appendix K to 10 CFR Part 50 did not require demonstration of the power measurement uncertainty and mandated a 2-percent margin, notwithstanding that the instruments used to calibrate the neutron flux instrumentation may be more accurate than originally assumed in the ECCS rulemaking. In the June 1, 2000, *Federal Register*, (Volume 65, pages 34913-34921), the Commission published a final rule to reduce an unnecessarily burdensome regulatory requirement by allowing licensees to justify a smaller margin for power measurement uncertainty by using more accurate instrumentation to calculate the reactor thermal power and thereby calibrate the neutron flux instrumentation.

Your application proposed changes to the SSES Unit 1 and 2 licenses and technical specifications to obtain a power uprate on the basis of plant modifications that would result in improved accuracy of the feedwater flow rate and feedwater temperature measurements used to calculate reactor thermal power. The improved instrumentation will allow operation of the SSES units with a reduced margin between the actual power level and the 102-percent margin used in the licensing basis ECCS analyses.

To complete its review of the proposed changes, the NRC staff requests a description of the programs and procedures that will control calibration of the Caldon LEFM and associated instrumentation that affect the total power uncertainty described in your power uprate application. Include in this discussion the procedures for:

1. Maintaining calibration,
2. Controlling software and hardware configuration,
3. Performing corrective actions,
4. Reporting deficiencies to the manufacturer, and
5. Receiving and addressing manufacturer deficiency reports.

Response

1. Maintaining calibration

The plant instruments that provide input into the heat balance are calibrated and maintained by either preventive maintenance activities and/or by surveillance activities. Instrumentation sensing the following parameters are input to the heat balance; reactor pressure, feedwater flow, CRD flow, feedwater temperature, recirculation pump power, Reactor Water Cleanup system temperature and flow, and total core flow.

Preventive Maintenance activities are defined as those activities that extend equipment service life or prevent equipment failure and are based on engineering judgement and manufacturers recommendations. Surveillance activities are those activities that are performed to satisfy Technical Specification or Technical Requirements Manual requirements.

For the subject instruments, loop calibrations are scheduled and performed in accordance with SSES "Routine Task System", "Surveillance Testing Program" and the "Maintenance and Control of Installed Instrumentation procedure. These programs and procedures are in accordance with SSES Section 17.2 "Quality Assurance During the Operation Phase".

2. Controlling software and hardware configuration

Controlling Software Configuration

The LEFM software configuration is controlled via a combination of processes that consists of the following:

- The PPL Susquehanna LLC Process Computer Software Quality Assurance program and referenced lower tier instructions to manage the software design, configuration, and control of Supplier services.
- The PPL Susquehanna LLC Modification process controls the system design, configuration changes, and installation.
- The PPL Susquehanna LLC Corrective action process and the Work order process is used to conform the system to it's design function.
- A unique LEFM Computer system SQA plan is written to prescribe any unique and additional processes used for this system.

The LEFM system was constructed under the auspices of the CALDON Quality Assurance program. The program is in compliance with Industry SQA standards and PPL's SQA program. Their internal program is used to control their development, verification, validation, and change control processes.

Controlling Hardware Configuration

PPL controls the hardware configuration of plant systems and components in accordance with a Configuration Management program that is pursuant to the SSES Section 17.2 "Quality Assurance During the Operation Phase". This program addresses the establishment and conformance with SSES design and licensing requirements, the SSES physical configuration, and associated documentation. These programs are applied to the equipment that affects the total power uncertainty described in our power uprate application.

3. Performing corrective actions

PPL implements a deficiency control program (Condition Report Process) that is focused on prompt identification, documentation and correction of conditions adverse to quality or safety. The program contains provisions for tracking and trending conditions and contains provisions for identifying and analyzing precursors to conditions adverse to quality. This program is pursuant to the SSES Section 17.2 "Quality Assurance During the Operation Phase" requirements. This program identifies and prioritizes the need for corrective actions. The corrective actions as deemed necessary are implemented in accordance with the appropriate plant programs. This program is applied to the equipment that affects the total power uncertainty described in our power uprate application.

4. Reporting deficiencies to the manufacturer

   Part/equipment deficiencies identified at SSES are documented using the Condition Report Process described above. The work group responsible for resolving the Condition Report will, as part of the investigation, contact the manufacturer as required.

   The Condition Report Process, includes process steps which require evaluation for reportability concerns. The reportability evaluation process includes the consideration for 10CFR21 reporting. This program is applied to the equipment that affects the total power uncertainty described in our power uprate application.

5. Receiving and addressing manufacturer deficiency reports.

   PPL implements a comprehensive Industry Event Review Program. The program's purpose is to collect lessons learned from the rest of the nuclear industry so to preclude similar events from occurring at SSES. Notices such as those received from the NRC, 10CFR21 reports, manufacturer / vendor notices, etc. are evaluated by the Industry Event Review Program (IERP).

   If the IERP determines that the notice is applicable to SSES, the Condition Report Process (described previously) is entered and utilized to control the evaluation, priority and tracking of any warranted corrective actions. This program is applied to the equipment that affects the total power uncertainty described in our power uprate application.

DRAFT FSAR MARKUPS

The safety/relief valve characteristic as modeled is shown in Figure 5.2-2 for the spring mode of operation. Typical valve characteristics are reflected in Figure 5.2-2A. The associated bypass, turbine control valve, and main steam isolation valve characteristics are also simulated in the model.

Closure time of the MSIVs is conservatively assumed to be less than or equal to the minimum closure time given in the Technical Specifications.

5.2.2.2.2 System Design

Reload specific evaluations are conducted to determine the required steam flow capacity of the safety/relief valves based on the following assumptions:

5.2.2.2.2.1 Operating Conditions

(1) operating power = 3510 MWt ~~(102% of rated power)~~, and

(2) vessel dome pressure ≤ 1050 psig, and

(3) core coolant flow = 108 million lbs/hr.

These conditions are the most severe because maximum stored energy exists at these conditions. At lower power conditions the transients would be less severe.

5.2.2.2.2.2 Transients

The overpressure protection system must accommodate the most severe pressurization transient. There are two major transients, the closure of all main steamline isolation valves and a turbine/generator trip with a coincident failure of the turbine steam bypass system valves that represent the most severe abnormal operational transient resulting in a nuclear system pressure rise. The evaluation of transient behavior with final plant configuration has shown that the isolation valve closure is slightly more severe when credit is taken only for indirect derived scrams. Therefore, it is used as the overpressure protection basis event. The cycle specific results are shown in figures 5.2-13 and 5.2-14 for Units 1 and 2 respectively. The peak pressures are determined for each of the components listed in section 5.2.2.2.1 and the minimum margin to their respective design limits is also determined. Calculated pressures are all within the respective acceptance criteria of 110% of the design pressure for the reactor pressure vessel and the reactor pressure boundary components. The feedwater piping connection has the smallest margin to its design limit during the transient. These margins are ~~25.2~~ 37.4 psi for Unit 1 Cycle ~~11~~ 12 and ~~36.9~~ 41.5 psi for Unit 2 Cycle ~~10.~~ 11. Table 5.2-9 lists the sequence of events of the various systems assumed to operate during the main steam line isolation closure with high neutron flux scram event for units 1 and 2.

5.2.2.2.2.3 Scram

The scram times assumed for the overpressure protection analysis are based on the maximum allowable values given in the Technical Specifications.

5.2.2.2.2.4 Safety/Relief Valve Transient Analysis Specifications

(1) valve groups: spring-action safety mode - 3 groups

(2) pressure setpoints: see Table 5.2-2

The setpoints are assumed at a conservatively high level above the nominal setpoints. This is to account for initial setpoint errors and any instrument setpoint drift that might occur during operation. The assumed setpoints in the analysis are 3% above the actual nominal setpoints. Conservative safety/relief valve response characteristics as shown in figure 5.2-6 are assumed.

For the analysis, the safety valves that were assumed to be out of service were those that had the lowest pressure setpoints. The assumed minimum number of operable S/RVs is in accordance with the Technical Specifications.

5.2.2.2.2.5 Safety Valve Capacity

Sizing of the safety valve capacity and the number of valves allowed to be out-of-service was based on assuring that the peak vessel pressure is less than the vessel code limit (1375 psig) in response to the reference transients Subsection 5.2.2.2.2.2. In addition, the analyses that are performed under Subsection 5.2.2.2.2.2 are also used to confirm that the capacity of the safety valves is adequate to assure that the component peak pressures during the transient are less than the limits listed in Subsection 5.2.2.2.1.

5.2.2.2.3 Evaluation of Results

5.2.2.2.3.1 Safety Valve Capacity

The required safety valve capacity is determined by analyzing the pressure rise from a MSIV closure with flux scram transient. The plant is assumed to be operating at the turbine-generator design conditions at a maximum vessel dome pressure equal to the maximum dome pressure allowed by Technical Specifications. The reactor power is assumed to be ~~102% of rated power~~ 3510 MWt. The analysis hypothetically assumes the failure of the direct MSIV position scram. The reactor is shut down by the backup, indirect, high neutron flux scram. The analysis indicates that the design valve capacity is capable of

TABLE 5.2-9

SEQUENCE OF EVENTS FOR MSIV ISOLATION CLOSURE
(SEE FIGURE 5.2-13)

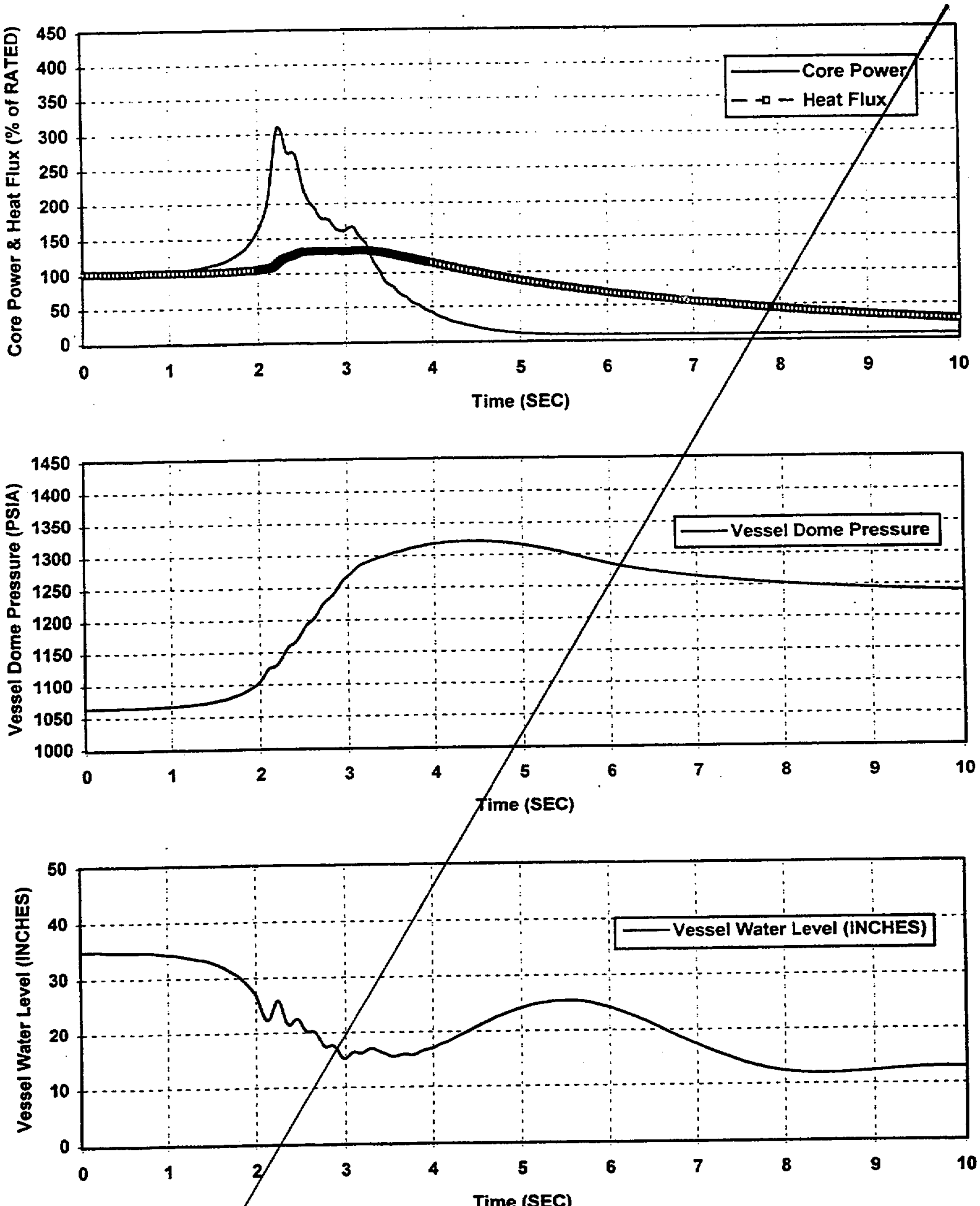
UNIT 1 CYCLE ~~11~~ 12

TIME-SEC	EVENT
0.00	Initiate closure of all main steam isolation valves (MSIV)
~~1.87~~ 1.86	Neutron flux reached the high APRM flux scram setpoint and initiated reactor scram
2.00	MSIV's completely closed
2.81	Steamline pressure reached the group safety valve pressure setpoint, and safety valves started to open.
~~4.60~~ 4.40	Time when peak pressure was reached in feedwater line penetration to reactor pressure vessel.

SEQUENCE OF EVENTS FOR MSIV ISOLATION CLOSURE
(SEE FIGURE 5.2-14)

UNIT 2 CYCLE ~~10~~ 11

TIME-SEC	EVENT
0.00	Initiate closure of all main steam isolation valves (MSIV)
~~1.81~~ 1.82	Neutron flux reached the high APRM flux scram setpoint and initiated reactor scram
2.00	MSIV's completely closed
~~2.81~~ 2.82	Steamline pressure reached the group safety valve pressure setpoint, and safety valves started to open.
~~4.40~~ 4.35	Time when peak pressure was reached in feedwater line penetration to reactor pressure vessel.



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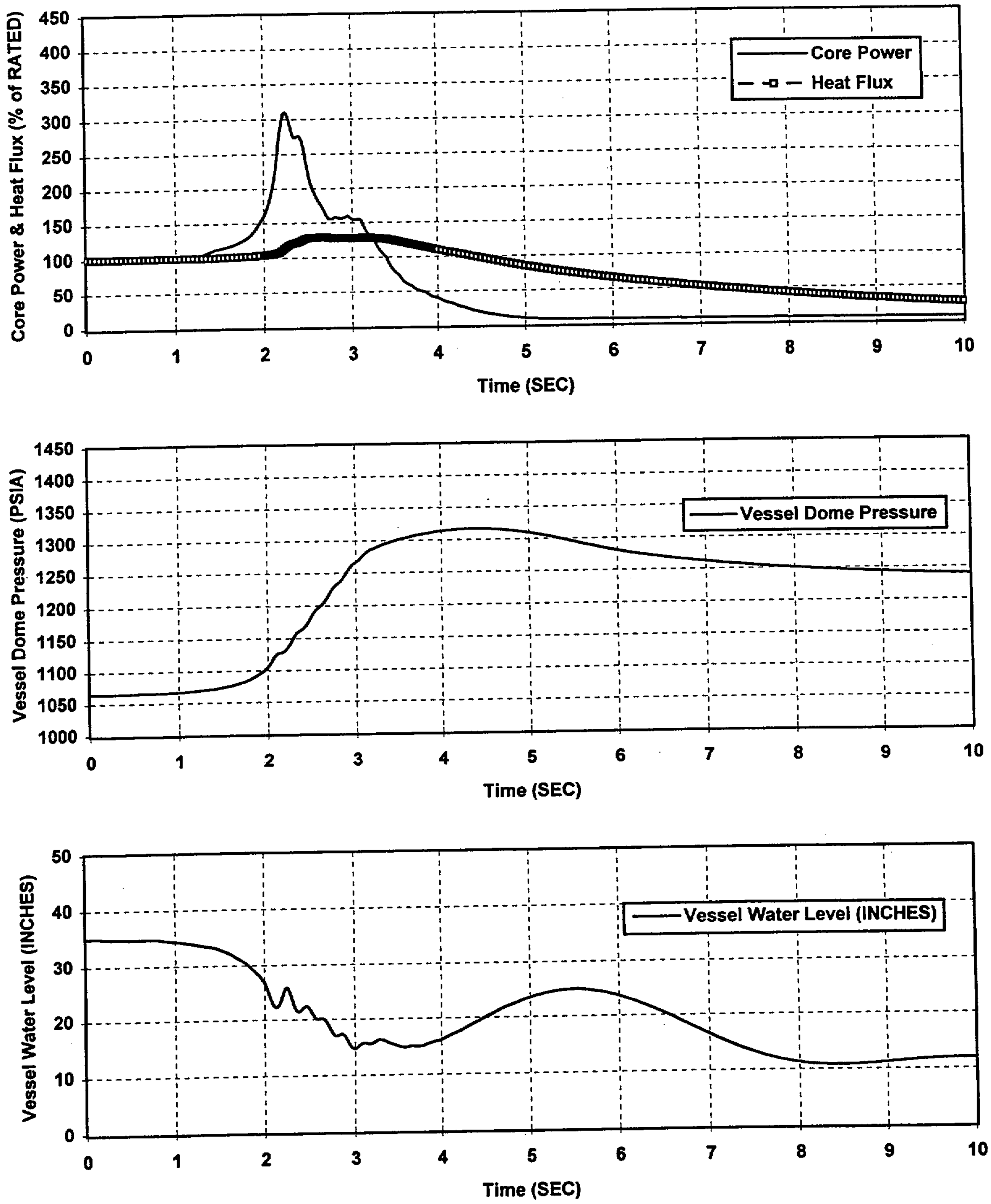
~~FSAR REV. 54, 10/99~~

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE ~~10~~ 11
FINAL SAFETY ANALYSIS REPORT

OVERPRESSURE PROTECTION ANALYSIS
(MSIV CLOSURE WITH HIGH FLUX SCRAM TRIP)
FSAR FIGURE 5.2-14

NUCLEAR FUELS



FSAR REV. 55

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 11
FINAL SAFETY ANALYSIS REPORT

OVERPRESSURE PROTECTION ANALYSIS
(MSIV CLOSURE WITH HIGH FLUX SCRAM TRIP)
FSAR FIGURE 5.2-14

NUCLEAR FUELS

For situations in which fuel damage is sustained, the extent of damage is determined by correlating fuel energy content, cladding temperature, fuel rod internal pressure, and cladding mechanical characteristics.

These correlations are substantiated by fuel rod failure tests and are discussed in Section 4.4 and Section 6.3.

15.0.3.3.2 Input Parameters and Initial Conditions for Analyzed Events

In general the limiting events analyzed within this section have values for input parameters and initial conditions as specified in Tables 15C.0-2 and 15D.0-2 for Units 1 and 2. These tables include the current conditions for power uprate. Analyses which assume data inputs different than the power uprate values are designated accordingly in the appropriate event discussion. Table 15E.0-2 provides the initial conditions used for the analysis of the non-limiting events for the initial cycle for Units 1 and 2.

15.0.3.3.3 Initial Power/Flow Operating Constraints

The analysis basis for most of the transient safety analyses is given in Tables 15C.0-2 and 15D.0-2 ~~the thermal power~~ at a core flow of 108 Mlbs/hr and a power ~~corresponding to 102% of the rated power.~~ However to assure that thermal margins are maintained over the entire power/flow operational space, the anticipated operational occurrences were analyzed over a range of power and flow conditions for the current cycles. In addition, single loop operation was analyzed for each of the anticipated operational occurrences and accidents. It was determined that for each anticipated event and the ASME overpressure analysis, the two loop results bound the results from single loop operation. Explicit analyses of LOCA and the pump seizure in single loop operation were also performed.

Figure 15E.0-1 is a typical power/flow map for a BWR. Figures 15C.0-1 and 15D.0-1 are the power/flow maps for the current cycles for Units 1 and 2.

Referring to Figure 15E.0-1, the apex of the bounded power/flow map is point A, the upper bound is the design flow control line (105%, rod line A-D'), the lower bound is the zero power line H'-J', the right bound is the rated pump speed line A-H', and the left bound is either the minimum pump speed line D-J or the natural circulation line D'-J'.

The power/flow map, A-D'-J'-H-A, represents the acceptable operational constraints for anticipated operational transient evaluations.

Any other constraint which may truncate the bounded power/flow map must be observed, such as the recirculation valve and pump cavitation regions, the licensed power limit and other restrictions based on pressure and thermal margin criteria. For instance, if the licensed power is 100% , the power/flow map is truncated by the line B-C on Figure 15E.0-1 and reactor operation must be confined within the boundary B-C-D'-J'-J-L-K-B. If the hydraulic behavior. Changes in the flow split between the bypass and active channel flow are accounted for during transient events.

e. Principal controller functions such as feedwater flow, recirculation flow, reactor water level, and pressure, are represented together with their dominant nonlinear characteristics.

f. The ability to simulate necessary reactor protection system functions is provided.

g. The control systems and reactor protection system models are described in detail in Reference 15.1-3

15.1.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15C.0-2 and 15D.0-2 for the current cycles for Units 1 and 2.

The transient model for the SSES Units 1 and 2 was initialized and executed for this event at one or more exposure steps for the current cycles. The initialization includes both the physics and thermal-hydraulic input that is exposure dependent. The Feedwater Controller Failure is analyzed for each of these exposures to determine the most limiting conditions for the cycles. The analyses are also performed over a range of power levels from 25% to ~~102%~~ 3510 MWt. The flow was held constant at 108 Mlbs/hr. In general, the limiting initial condition for this event is full flow of 108 Mlbs/hr. If there is reason to believe that the limiting initial flow condition is other than full flow, additional analyses are performed at lower flows.

The analyses also consider the following:

1. Steam bypass and Recirculation Pump Trip operable,

2. Steam bypass inoperable and Recirculation Pump Trip operable,

3. Steam bypass operable and Recirculation Pump Trip inoperable.

4. Realistic Scram Insertion Time or Maximum Allowable Scram Insertion Time.

The analysis is performed using relief/safety valve setpoints corresponding to the "relief mode" since the ΔCPR's are more limiting under this condition.

The initiating event for this transient is the failure of the feedwater control system causing a step change of feedwater flow from its initial steady-state value to the maximum value of full power feedwater flow.

15.1.4.2.2 Systems Operation

In this transient, the core performance analysis assumes normal functioning of plant instrumentation and controls, specifically the pressure regulator and level control systems.

Additionally, normal operation of relief valve discharge line temperature sensors and the suppression pool temperature sensors provides operator information as the basis for initiating a timely plant shutdown.

15.1.4.2.3 The Effect of Single Failures and Operator Errors

Failure of additional components (e.g., pressure regulator, feedwater flow controller) is discussed elsewhere in Chapter 15. In addition, a detailed discussion of such effects is given in Appendix 15A.

15.1.4.3 Core and System Performance

15.1.4.3.1 Mathematical Model

It was determined that this event is not limiting from a core performance standpoint. Therefore a qualitative presentation of results is described below.

15.1.4.3.2 Input Parameters and Initial Conditions

of 3510 MWt

It is assumed that the reactor is operating at an initial power level ~~corresponding to 102 % of rated steam flow conditions~~ when a safety/relief valve is inadvertently opened. Manual recirculation flow control is assumed. Flow through the valve at normal plant operating conditions stated above is approximately 928,800 lbs/hr.

15.1.4.3.3 Qualitative Results

The opening of a safety/relief valve allows steam to be discharged into the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient. The pressure regulator senses the nuclear system pressure decrease and within a few seconds closes the turbine control valve far enough to stabilize reactor vessel pressure at a slightly lower value and reactor power settles at nearly the initial power level. Thermal margins decrease only slightly through the transient, and no fuel damage results from the transient. MCPR is essentially unchanged and therefore the safety limit margin is unaffected.

Continued maximum steam flow to the suppression pool will be terminated by operator action.

Operation of safe shutdown features, though not included in this simulation, is expected to be utilized to maintain adequate water level.

15.3.3.2.3 The Effect of Single Failures and Operator Errors

Single failures in the scram logic originating via the high vessel level (L8) trip are similar to the considerations in Subsection 15.3.1.2.3.2.

15.3.3.3 Core and System Performance

15.3.3.3.1 Mathematical Model

The pump seizure accidents from single loop and two loop operation were analyzed using the methods and model described in References 15.3-1 and 15.3-6 through 15.3-11.

15.3.3.3.2 Input Parameters and Initial Conditions

For the purpose of evaluating consequences to the fuel thermal limits, this transient event is assumed to occur as a consequence of an unspecified, instantaneous stoppage of one recirculation pump shaft.

The analysis for pump seizure from single loop operation was performed for an initial power level of approximately 76% ~~(power uprate)~~ and 52 Mlbs/hr core coolant flow. For the analysis with two loop operation, the initial conditions for Units 1 and 2 are given in ~~Tables 15C.0-2 and 15D.0-2.~~ Figures 15C.3.3-1 and 15D.3.3-1

Also, the reactor is assumed to be operating at thermally limited conditions for each of the initial conditions analyzed.

15.3.3.3.3 Results

Figures 15C.3.3-1, 15D.3.3-1, 15C.3.3-2 and 15D.3.3-2 present the results of the accident. Core coolant flow drops rapidly. The MCPR decrease is significant and can lead to violation of the MCPR safety limit. To ensure that this event does not violate the criteria described in Subsection 15.0.3.1.2, Unacceptable Results for Infrequent Incidents (Abnormal (Unexpected) Operational Transients), the MCPR prior to pump seizure was determined and set such that the accident would yield less than 10% of the dose permitted by 10CFR100. This initial MCPR operating limit was determined for both single loop and two loop operation. These MCPR operating limits are given in Table 15C.0-4 for Unit 1 and in Table 15D.0-4 for Unit 2 for single and two loop operation.

15.3.3.5.1 Design Basis Analysis

To determine the radiological consequences, the following assumptions were used to calculate the number of rods which experience boiling transition during the pump seizure accident:

1. The number of rods in boiling transition is assumed to be equal to the number of rods that would be in boiling transition based on steady state operation at the minimum CPR during the event. All rods which experience boiling transition are assumed to fail. This is a very conservative assumption for a pump seizure accident because the minimum CPR occurs for such a short period of time.

2. The power distributions determined to be limiting in MCPR safety limit analyses were used in this analysis.

The following assumptions were used to evaluate the dose from this accident:

1. The source terms were based on reactor operation at 3616 MWt ~~105% of uprated~~ power conditions.

2. 10% of the iodine isotopes and noble gases available in the fuel are released from the damaged rods. A 1.5 multiplier is used to account for power peaking.

3. 10% of the iodine isotopes and 100% of the noble gases released from the fuel are transported into the steamline.

4. 10% of the iodine and 100% of the noble gases transported into the steamline are assumed to remain airborne in the condenser.

5. The condenser is assumed to have a leak rate of 1% per day.

6. Leakage from the core, into the reactor building and out to the environment through the SGTS is negligible in comparison to the leakage rate through the condenser.

Tables 15C.3.3-3 and 15C.3.3-4 provide the activity released per failed rod and the calculated airborne activity in the condenser (Ci/failed rod) for the two loop and single loop pump seizure event for Unit 1 with ATRIUM™-10 and 9x9-2 fuel, respectively. Table 15D.3.3-3 provides the activity released per failed rod and the calculated airborne activity in the condenser (Ci/failed rod) for the two loop and single loop pump seizure event for Unit 2.

Tables 15C.3.3-5 and 15C.3.3-6 provide the activity released per failed rod to the environs for the two loop and single loop pump seizure event for Unit 1 with ATRIUM™-10

15.4.1.1.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since it is a highly localized event and does not result in any change in the core pressure or temperature.

15.4.1.1.5 Radiological Consequences

An evaluation of the radiological consequences was not made for this event since no radioactive material is released from the fuel.

15.4.1.2 Continuous Rod Withdrawal During Reactor Startup

15.4.1.2.1 Identification of Causes and Frequency Classification

The event is defined as: while operating below the low power setpoint and coincident with a failure or bypass of the RWM and/or RSCS, the operator makes a procedural error and withdraws an out of sequence control rod of maximum worth. The probability of initial causes or errors of this event alone is considered low enough to warrant its being categorized as an infrequent incident. The probability of further development of this event is extremely low because it is contingent upon the failure of the RWM system and/or the RSCS, concurrent with a high worth rod, out-of-sequence rod selection contrary to procedures, plus operator ignorance of any alarm annunciations prior to safety system actuation.

It is possible but highly unlikely that the RWM and the RSCS could be bypassed at the same time. However, whenever the RWM is inoperable or bypassed, there is a Technical Specification requirement that a second operator verify that the correct control rod withdrawal sequence is followed.

15.4.1.2.2 Sequence of Events and Systems Operation

15.4.1.2.2.1 Sequence of Events

Control rod withdrawal errors are not considered credible in the startup and low power ranges. The RWM plus procedural requirements prevent the operator from selecting and withdrawing an out-of-sequence control rod. In addition, the RSCS would also prevent the operator from selecting and withdrawing an out-of-sequence control rod.

The purpose of the RWM is to control rod patterns during startup, such that only specified rod sequences and relative positions are allowed over the operating range from all control rods inserted to approximately 10% of rated core power. The sequences effectively limit the potential amount and rate of reactivity increase during a Control Rod Drop Accident. The RWM is designed to act as a backup to operator control of the rod sequences. Therefore if the

RWM is inoperable or bypassed the Technical Specifications require that a second operator verify that any subsequent rod selection and withdrawal is in accordance with the specified rod sequence.

Continuous control rod withdrawal errors during reactor startup are precluded by the RSCS. The RSCS is a backup to the RWM and prevents the withdrawal of an out-of-sequence control rod in the 100% to 75% control rod density range and limits rod movement to the banked position mode of rod withdrawal from the 75% rod density to the preset power level. Since only in-sequence control rods can be withdrawn in the 100% to 75% control rod density and control rods are withdrawn in the banked position mode from the 75% control rod density point to the preset power level, there is no basis for the continuous control rod withdrawal error in the startup and low power range. The low power range is defined as zero power to the RSCS low power setpoint, i.e., 10% of rated core power. For RWE above low power setpoint see Subsection 15.4.2. The banked position mode of the RSCS is described in Reference 15.4-2. approximately

In the unlikely event that both the RWM and RSCS fail to prevent an out-of-sequence control rod from being withdrawn in the reactor startup range, fuel failure will not occur as shown by generic analyses performed by General Electric in Reference 15.4-10. Protection is provided by the IRM upscale scram function and/or APRM scram which are both single failure proof designed systems.

15.4.1.2.2.2 Identification of Operator Actions

No operator actions are required to preclude this event since the plant design as discussed above prevents its occurrence.

15.4.1.2.2.3 Effects of Single Failure and Operator Errors

If any one of the operations involved the initial failure or error followed by another SEF or SOE, the necessary safety actions are taken (e.g., rod blocks) prior to any limit violation.

15.4.1.2.3 Core and System Performance

The performance of the RWM and RSCS prevent erroneous selection and withdrawal of an out-of-sequence control rod. The core and system performance is not affected by such an operator error.

No mathematical models are involved in this event. The need for input parameters or initial conditions is not required as there are no results to report. Consideration of uncertainties is not appropriate.

15.4.5.2.3 The Effect of Single Failures and Operator Errors

This transient leads to a gradual rise in reactor power level. Corrective action occurs from either the high flux trip or the high pressure trip and, being part of the reactor protection system, these trips are designed to meet the single failure criteria. Therefore, shutdown is assured. Operator errors are not of concern here in view of the fact that automatic shutdown events follow soon after the postulated failure.

15.4.5.3 Core and System Performance

15.4.5.3.1 Mathematical Model

The nonlinear dynamic model described in Reference 15.4-12 is used to simulate this event.

15.4.5.3.2 Input Parameters and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions tabulated in Tables 15C.0-2 and 15D.0-2 for Units 1 and 2.

For this event a number of different Power/Flow conditions are analyzed. The initial conditions that are typically examined are 51% ~~52%~~P/37Mlbs/hr, 68% ~~69%~~P/60Mlbs/hr and 86% ~~87%~~P/85Mlbs/hr. Analyses are performed using the Technical Specification maximum allowable scram insertion times. Analyses are performed for the steam bypass system operable and for it inoperable. The ramp rate of 0.25%, at which the recirculation pump speed is assumed to increase, is adjusted slightly for each initial Power/Flow condition to achieve, as nearly as possible, simultaneous scrams on high neutron flux and high pressure. The purpose of this assumption is to minimize the thermal margin for this event.

The high neutron flux trip and the high pressure trip are set at their analytical limits. The safety/relief valves are set to open based on their nominal relief valve setpoints.

15.4.5.3.3 Results

Figures 15C.4.5-1 and 15D.4.5-1 show the results of one of the transients examined for Units 1 and 2 for the current cycles. The nuclear system pressure increase is limited by the high pressure analytical trip setpoint and operation of the safety/relief valves which are set to open at the nominal relief valve setpoints.

The peak neutron flux rise approaches the high neutron flux analytical trip setpoint. Since the transient is relatively slow, the change in heat flux is essentially the same as the change in neutron flux.

b) An average of 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity in a failed fuel rod is assumed to be released. These percentages are consistent with actual measurements made during defective fuel experiments (Reference 15.4-9).

c) The fraction of solid fission product activity available for release from the fuel is negligible.

d) The fission products produced during the nuclear excursion are neglected.

The following assumptions are used in calculating the amount of fission product activity transported from the reactor vessel to the main condenser:

a) The recirculation flow rate is approximately 25 percent of rated, and the steam flow to the condenser is 5 percent of rated. The 25 percent recirculation flow and approximately 5 percent steam flow are the maximum flow rates compatible with the maximum fuel damage. The approximately 5 percent steam flow rate is greater than that which would be in effect at the reactor power level assumed in the initial conditions for the accident. This assumption is conservative because it results in the transport of more fission products through the steamlines than would be expected. Because of the relatively long fuel-to-coolant heat transfer time constant, steam flow is not significantly affected by the increased core heat generation within the time required for the main steamline isolation valves to achieve full closure.

b) The main steamline isolation valves are assumed to receive an automatic closure signal 0.5 second after detection of high radiation in the main steamlines and to be fully closed at 5 seconds from the receipt of the closure signal. The automatic closure signal originates from the main steamline radiation monitors. The total time required to isolate the main steamlines (5.5 seconds) combined with the assumptions, dictates the total amount of fission product activity transported to the condenser before the steamlines are isolated.

c) All of the noble gas activity is assumed to be released to the steam space of the reactor vessel.

d) The mass ratio of the halogen concentration in steam to that of the water is assumed to be 2 percent.

e) Fission product plateout is neglected in the reactor vessel, main steam lines, turbine, and condenser.

Of those fission products released from the fuel and transferred to the condenser, it is assumed that 100 percent of the noble gases are airborne in the condenser. The iodine activity airborne in the condenser is a function of the partition factor, volume of air, and volume of water. A partition factor of 140 is assumed in condenser for iodine activity. By

(6) The energy required to produce cladding failure due to compression for an SPC ATRIUM™-10 fuel rod is approximately 216 ft-lbs. This is based upon a 1% plastic hoop strain in the rod.

15.7.4.3.3 Results

The results for fuel handling and equipment handling accidents involving freshly used discharged ATRIUM™-10 fuel are the most limiting of all the fuel types ~~currently in~~ Susquehanna Units 1 and 2. ~~The ATRIUM™-10 results bound the SPC 9x9-2 assembly, and all current lead fuel assembly types including the GE12 and SVEA-96+ designs.~~ The ATRIUM™-10 results also conservatively bound all fuel designs in the spent fuel pool, including the SPC 8x8, ~~and~~ GE 8x8, ~~assemblies.~~ SPC 9x9-2, GE 12, and SVEA-96+ designs.
For each fuel type~~s~~(s) specified below, the basis for why ATRIUM™-10 is more limiting is provided:

GE 8x8 and SPC 8C8 fuel

Due to the extended decay time these fuel bundles have experienced since discharge from the reactor, the ATRIUM™-10 source term will be larger.

GE12 and ABB SVEA-96+ LUAs

See section 15.7.4.3.3.3.3 for fuel handling accident results. The number of LUA fuel assembly failures, as calculated by the respective fuel vendors, is less than ATRIUM™-10. The same conclusion can be extended to the equipment handling accident.

SPC 9x9-2 fuel

SPC has reported and documented that the ATRIUM™-10 fuel and equipment handling accident is bounding over the 9x9-2. This is reasonable because the threshold to fail one ATRIUM™-10 fuel assembly is less than that to fail one 9x9-2 fuel assembly. Since the source term is about the same for an ATRIUM™-10 and 9x9-2 fuel assembly, the ATRIUM™-10 fuel and equipment handling accidents would result in more assembly failures and a higher radiological release.

15.7.4.3.3.1 Energy Available

For the initial impact of the fuel handling accident, a load of 1500 lbs. representing the channeled fuel assembly, grapple, and mast is assumed to drop 32.95 ft and impact onto other fuel assemblies with a maximum kinetic energy of 49,425 ft-lbs. Following the initial impact, the fuel assembly, grapple, and mast are assumed to tip over and impact horizontally with a maximum kinetic energy of approximately 17,272 ft-lbs.

fuel assemblies. Thus, the impact of the equipment handling accident yields the following fuel failures:

Struck Assemblies	366 rods

15.7.4.3.3.3.2 Second Impact Failures

Following the initial impact in the fuel handling accident, the fuel assembly, grapple, and mast are assumed to tip over and impact horizontally with a maximum kinetic energy of approximately 17,272 ft-lbs.

Half of that energy is assumed to be absorbed by the non-fuel components of the struck assemblies. The ATRIUM™-10 assembly non-fuel components cladding weight fraction is 0.478. Therefore the total amount of energy absorbed by the struck fuel rods is approximately 4128 ft-lbs. Dividing this by the cladding failure threshold of 216 ft-lbs yields approximately 19 failed rods in the struck fuel assemblies. Thus, the second impact of the fuel handling accident yields the following fuel failures:

Struck Assemblies	19 rods (2nd Impact)

15.7.4.3.3.3.3 Total Failures

The total number of failed rods resulting from the fuel handling accident is as follows:

First impact	146 rods
Second impact	19 rods
	165 total failed rods (1.88 assemblies)*

The total number of failed rods resulting from the equipment handling accident is as follows:

First impact	366 total failed rods (4.17 assemblies)*

~~Susquehanna Unit 1 contains~~ The four ABB SVEA-96+ lead fuel assemblies and ~~Susquehanna Unit 2 contains~~ four GE12 lead fuel assemblies have been discharged from the Susquehanna Unit 1 and Unit 2 reactors. For the GE12 lead fuel assembly design, GE determined that 151 fuel rods (1.64 assemblies) would fail as a result of the fuel handling accident (Reference 15.7-6). For the SVEA-96+ lead fuel assembly design, ABB determined that 124 fuel rods (1.29 assemblies) would fail as a result of the fuel handling accident (Reference 15.7-7). The results for both lead fuel assembly designs are bounded by the ATRIUM-10 fuel handling accident results (1.88 failed assemblies).

However, to conservatively address the issue of lead fuel assemblies, radiological dose results are included in Table 15.7-16A which assume that another ATRIUM™-10

15.8.1.5 Mathematical Models

15.8.1.5.1 Power Uprate Analysis

The ATWS analysis for power uprate was performed by General Electric, and the analysis methods are described in Section 2.0 of GENE-637-024-0893.

15.8.1.5.2 Post-Power-Uprate Analyses

For post-power-uprate fuel cycles, assessment of Susquehanna performance for ATWS events is performed in-house by PP&L when changes in reactor operating conditions or changes in core design warrant reanalysis (Ref. 15.8-7). Only the limiting ATWS events, MSIV Closure and Pressure Regulator Failure - Open, are reevaluated. Calculation of peak vessel pressure is performed with the RETRAN computer code. This is the same model which is used to perform the nuclear fuel reload analysis. An assessment for PCT is made by comparing the core-average heat flux calculated with RETRAN against heat flux response predicted by General Electric in GENE-637-024-0893.

The PP&L SABRE code is used to evaluate the peak suppression pool temperature. SABRE results have been used by the NRC in evaluating reactor water level control strategies for ATWS mitigation, and the NRC has concluded that SABRE predictions for ATWS scenarios are comparable to results obtained with TRAC-BF1 (with 1-D neutronics) and RAMONA-4B (Ref. 15.8-8).

15.8.1.6 Input Parameters and Initial Conditions

Input parameters for the ATWS analysis are listed in Tables 15.8-1 and 15.8-2. The Hot Shutdown Boron Concentration and the Hot Shutdown Reactivity are based on Hot Full Power Xenon concentration.

~~In accordance with GENE 637-024-0893,~~ The ATWS simulations are initiated at a core power of 100% of rated or greater ~~3441 MWth,~~ and 87 MLbm/hr total core flow. This power/flow condition corresponds to the Extended Load Line Limit (ELLL). The cycle exposure corresponds to end of full power (all rods out). Tables 15.8-3 and 15.8-4 list the initial conditions.

15.8.2 Inadvertent Control Rod Withdrawal

In Section 3.1.16 of NEDE-24222 (Ref. 15.8-9), General Electric presents a detailed discussion of the consequences of a rod withdrawal error at full power and within the startup range. GE has concluded that the consequences of the control rod withdrawal error are such that analysis of this event is not necessary.

TABLE 15.8-2

INPUT PARAMETERS FOR UNIT 2 CYCLE ~~10~~ 11 ATWS ~~CYCLE 10~~ ANALYSIS

Closure Time of MSIV (sec)	4
ATWS High Pressure RPT Setpoint, UAL (psig)	1200
Setpoint for Low Water Level Closure of MSIV	L1(-129")
Setpoint for Low Steam Line Pressure Closure of MSIV (psig)	861
Relief Valve Setpoints	†
HPCI Flow Rate (gpm)	5000
HPCI Start/Stop Levels	L2(-38") / L8(+54")
RCIC Flow Rate (gpm)	600
RCIC Start/Stop Levels	L2(-38") / L8(+54")
Hot Shutdown Boron Weight (ppm)	494
~~Hot Shutdown Reactivity ($)~~	~~11.4~~
Nominal SLCS Boron Injection Rate (Lbm/sec)	0.28
Boron Transport Time from SLCS Pumps to Vessel (sec)	30
Condensate Storage Tank Water Temperature (°F)	123
Steam Condensation Efficiency on Subcooled Makeup Flow When Spargers are Uncovered (%)	95
ATWS Low Water Level RPT Setpoint	-38" ~~-45"~~
RHR Pool Cooling Capacity (2 Loops) (Btu/sec °F)	630
Service Water Temperature (°F)	88

† SRV set points are taken from Table 1 of GENE-637-024-0893, Supplement 1.

TABLE 15.8-4

INITIAL OPERATING CONDITIONS FOR UNIT 2 CYCLE 11 ~~10~~ ATWS ANALYSIS

Dome Pressure (psia)	†
Total Core Flow (Mlbm/hr)	87
Core Thermal Power (Mwth)	3489 ~~3441~~
Narrow Range Water Level (inches)	+35[#]
Suppression Pool Liquid Volume (ft^3)	122,986[‡]
Suppression Pool Temperature (°F)	90

† A pressure of 1053 psia was used for the suppression pool temperature calculation (SABRE model), and a pressure of ~1050 psia was used for the peak pressure and PCT calculation (RETRAN model). Initial dome pressure is code-calculated in Susquehanna RETRAN model based on other specified input parameters.

[#] Although +35" is used in the analysis, an initial Narrow Range level any where in the range +33" to +37" is acceptable.

‡ Although this volume is slightly higher (0.17%) than the value used by GE in GENE-637-024-0893 it corresponds to a suppression pool level of 22 feet which is the minimum value allowed by Technical Specifications.

TABLE 15.8-6

UNIT 2 CYCLE ~~10~~ 11 SEQUENCE OF EVENTS FOR MSIV CLOSURE ATWS

Event	Time (sec)
Nominal 4-second MSIV closure is initiated—scram fails.	0
Peak neutron flux (~~376~~ 385%) occurs.	4.4 ~~4.2~~
ATWS High Pressure Setpoint (1200 psig) is reached—RPT is initiated.	5.0
Peak vessel pressure is reached (~~1333.9~~ 1341.1 psig).	8.2 ~~8.0~~
SLCS is initiated by operator.	95
Feedwater flow begins to decline because of low steamline pressure.	98
HPCI and RCIC injection begins.	140
Operator begins to throttle high pressure injection to reduce level to within level control band defined by EOP.	300 ~~250~~
RHR flow begins (Suppression Pool Cooling).	1,000
Hot Shutdown Boron Weight is injected—operator initiates restoration of reactor water level.	1,535
Peak suppression pool temperature (~~176.2~~ 179.8 °F) occurs.	~~1,937~~ 1,727

TABLE 15.8-8

UNIT 2 CYCLE ~~10~~ 11 RESULTS FOR MSIV CLOSURE ATWS EVENT

Parameter	Result	Limit
Peak Vessel Pressure (psig)†	~~1333.9~~ 1341.1	1500
Peak Clad Temperature (°F)†	<1463	2200
Peak Suppression Pool Temperature (°F)	~~176.2~~ 179.8	190

† One SRV is out of service.

TABLE 15.8-10

UNIT 2 CYCLE ~~10~~ 11 SEQUENCE OF EVENTS FOR PREGO ATWS

Event	Time (sec)
Pressure Regulator Fails to Maximum Demand	0
Pressure and Power Begin to Fall	0
MSIVs closure on low steamline pressure—scram fails	7.7 ~~7.3~~
Peak neutron flux (~~430%~~ 442) occurs	12.3 ~~11.9~~
ATWS High Pressure Setpoint (1200 psig) reached—RPT is initiated	14.3 ~~13.9~~
Vessel pressure peaks (~~1301.3~~ 1306.1 psig)	15.8 ~~15.3~~
Feedwater capacity begins to decline	14 ~~103~~
SLCS Initiated by Operator	106 ~~113~~
HPCI and RCIC Injection Begin	38 ~~147~~
Operator begins to throttle high pressure injection to reduce level to within level control band defined by EOP.	300 ~~250~~
RHR Flow Begins (Suppression Pool Cooling)	1000
Hot Shutdown Boron Weight is injected—operator initiates restoration of reactor water level	1546 ~~1553~~
Peak suppression pool temperature (~~177.1~~ 166.6 °F) is reached	2040 ~~1813~~

TABLE 15.8-12

UNIT 2 CYCLE ~~10~~ 11 RESULTS FOR PREGO ATWS EVENT

Parameter	Result	Limit
Peak Vessel Pressure (psig)	1306.1 ~~1301.3~~	1500
Peak Clad Temperature (°F)	<1463	2200
Peak Suppression Pool Temperature (°F)	~~177.1~~ 166.6	190

TABLE 15D.0-1
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 2 CYCLE ~~10~~ 11

Section	Figure	Description[1]	Maximum Neutron Flux,% of Rated	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam line Pressure psig	Maximum Core Average Surface Heat Flux, %	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.1		DECREASE IN REACTOR COOLANT TEMPERATURE									
15.1.1		Loss of Feedwater Heater	NOTE 5	NOTE 5	NOTE 5	NOTE 5	NOTE 5	~~.18~~ 0.17	a	0	0 sec
15.1.2	15D.1.2-1	Feedwater Controller Failure (94% Power, 108 Mlb$_m$/hr, Realistic Scram Time)	~~183~~ 172	~~1141~~ 1148	~~1170~~ 1178	~~1150(4)~~ 1156(4)	~~111~~ 110	0.27	a	13	5 ~~10~~ sec (estimate)
15.1.3	15D.1.3-1	Pressure Regulator Failure - Open	102	1106	1129	1106	103	0.01	a	2	See Text
15.1.4		Inadvertent Opening of Safety or Relief Valves	See Text						a		
15.1.6		RHR Shutdown Cooling Malfunction	See Text						a		
15,2		INCREASE IN REACTOR PRESSURE									
15.2.1		Pressure Regulator Failure – Closed	See Text						a		
15.2.2		Generator Load Reject – Bypass Operable	See Text and Appendix 15E						a		
15.2.2	15D.2.2-1	Generator Load Reject- Without Bypass (~~102%~~ 100.6% Power, 108 Mlb$_m$/hr, Realistic Scram Time)	~~295~~ 265	~~1174~~ 1173	~~1201~~ 1199	~~1224~~ 1222	~~120~~ 115	~~0.35~~ 0.30	a	16	20 sec estimate
15.2.3		Turbine Trip - Bypass Operable	See Text and Appendix 15E						a		
15.2.3	15D.2.2-1	Turbine Trip – Without Bypass (~~102%~~ 100.6% Power, 108 Mlb$_m$/hr, Realistic Scram Time)	~~295~~ 265	~~1174~~ 1173	~~1201~~ 1199	~~1224~~ 1222	~~120~~ 115	~~0.35~~ 0.30	a	16	20 sec estimate
15.2.4		Inadvertent MSIV Closure	See Text and Appendix 15E						a		

TABLE 15D.0-1
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 2 CYCLE ~~10~~ 11

Section	Figure	Description[1]	Maximum Neutron Flux,% of Rated	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam line Pressure psig	Maximum Core Average Surface Heat Flux, %	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.2.5		Loss of Condenser Vacuum	See Text and Appendix 15E						Moderate		
15.2.6		Loss of Auxiliary Power Transformer	See Text and Appendix 15E						Moderate		
15.2.6		Loss of All Grid Connections	See Text and Appendix 15E						Moderate		
15.2.7		Loss of All Feedwater Flow	See Text and Appendix 15E						Moderate		
15.2.8		Feedwater Piping Break	See Section 15.6.6								
15.2.9		Failure of RHR Shutdown Cooling	See Text								
15.3		<u>DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE</u>									
15.3.1		Trip of One Recirculation Pump Motor	See Text and Appendix 15E						Moderate		
15.3.2		Trip of Both Recirculation Pump Motors	See Text and Appendix 15E						Moderate		
15.3.3		Seizure of One Recirculation Pump (Single Loop Operation)	76	1135 ~~1142~~	1148 ~~1157~~	1145 ~~1160~~	76	0.75 ~~1.70~~	Limiting Fault		
15.3.4		Recirculation Pump Shaft Break	See Text						Limiting Fault		

TABLE 15D.0-1
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 2 CYCLE ~~10~~ 11

Section	Figure	Description[1]	Maximum Neutron Flux,% of Rated	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam line Pressure psig	Maximum Core Average Surface Heat Flux, %	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.4		REACTIVITY AND POWER ANOMALIES									
15.4.1.1		RWE – Refueling	See Text						Infrequent		
15.4.1.2		RWE – Startup	See Text						Infrequent		
15.4.2		RWE - At Power, 108 Mlbs/hr, Bypass Operable	See Text	Note 5	Note 5	Note 5	Note 5	~~0.32~~ 0.30	Moderate		
15.4.3		Control Rod Maloperation	See Subsections 15.4.1 and 15.4.2								
15.4.4		Startup of Idle Recirculation Loop	See Text and Appendix 15E						Moderate		
15.4.5	15D.4.5-1	Recirculation Flow Controller Failure[3]	122	1107	~~1146~~ 1147	~~1059~~ 1107	118	~~0.38~~ 0.44	Moderate	2	~~3~~ 4 sec
15.4.7		Misplaced Bundle Accident	See Text	Note 5	Note 5	Note 5	Note 5	~~0.22~~ 0.19	Infrequent		
15.4.7		Rotated Bundle Accident	See Text	Note 5	Note 5	Note 5	Note 5	~~0.36~~ 0.31	Infrequent		

TABLE 15D.0-1
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 2 CYCLE ~~10~~ 11

Section	Figure	Description[1]	Maximum Neutron Flux,% of Rated	Maximum Dome Pressure psig	Maximum Vessel Pressure psig	Maximum Steam line Pressure psig	Maximum Core Average Surface Heat Flux, %	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.5		INCREASE IN REACTOR INVENTORY									
15.5.1		Inadvertent HPCI Pump Start	See Text and Appendix 15E						Moderate		
15.5.3		BWR Transients That Increase Reactor Coolant Inventory	See Sections 15.1 and 15.2								

Notes

1. Unless otherwise stated, the plant initial condition listed in this table for transients is: 102% Power, 108 Mlbs/hr Flow, EOC-Reactor Pump Trip Operable, Bypass Operable, Realistic Scram Time.
2. Minimum MCPR operating limit for Single Loop Operation, see Text.
3. Recirculation Flow Controller Failure transients are initiated from low power/low flow conditions. This one started at 68 ~~60~~% Power and 60 Mlbs/hr flow.
4. Steam line pressure is at the turbine stop valve for events in which the turbine trips. For other transients the steam line pressure is assumed to be no higher than the reactor vessel dome pressure.
5. These Anticipated Operational Occurrences are analyzed as steady-state events.

TABLE 15D.0-1A

SUMMARY OF ACCIDENTS
UNIT 2 CYCLE ~~10~~ 11

SUBSECTION I.D.	TITLE	FAILED FUEL RODS	
		REALISTIC UTILITY ASSUMPTIONS	WORST CASE NRC ASSUMPTIONS
15.3.3	SEIZURE OF ONE RECIRCULATION PUMP TWO LOOP OPERATION SINGLE LOOP OPERATION	 NONE NONE	 ~~7681~~ 7741 ~~7621~~ 7986
15.3.4	RECIRCULATION PUMP SHAFT BREAK	SEE 15.3.4 TEXT	
15.4.9	ROD DROP ACCIDENT	~~≤890~~ ≤1000	1000
15.6.2	INSTRUMENT LINE BREAK	NONE	NONE
15.6.4	STEAM SYSTEM PIPE BREAK OUTSIDE CONTAINMENT	NONE	NONE
15.6.5	LOCA WITHIN RCPB	NONE	100%
15.6.6	FEEDWATER LINE BREAK	NONE	NONE
15.7.1.1	MAIN CONDENSER GAS TREATMENT SYSTEM FAILURE	NONE	NONE
15.7.3	LIQUID RADWASTE TANK FAILURE	N/A	N/A
15.7.4	FUEL HANDLING ACCIDENT EQUIPMENT HANDLING ACCIDENT	165* 366*	165* 366*
15.7.5	CASK DROP ACCIDENT	NONE	---

*Not Unit Specific, see 15.7.4

TABLE 15D.0-2

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS
UNIT 2 CYCLE ~~10~~ 11

		~~POWER UPRATE~~
1.	Thermal Power Level, MWT Rated Value Analysis Value	3489 ~~3441~~ (100%) 3510 (~~102%~~) 100.6%
2.	Steam Flow, Mlbs/hr (At 100% Power and 108 Mlbs/hr)	~~14.184~~ 14.415
3.	Maximum Core Flow, Mlbs/hr	108.0(3)
4.	Feedwater Flow Rate, Mlbs/hr (At 100% Power and 108 Mlbs/hr)	~~14.152~~ 14.383
5.	Feedwater Temperature,°F (At 100% Power and 108 Mlbs/hr)	~~387.1~~ 388.7
6.	Vessel Dome Pressure,psig (At 100% Power and 108 Mlbs/hr)	~~1035.3~~ 1034.5
7.	Vessel Core Pressure,psig at Channel Exit (At 100% Power and 108 Mlbs/hr)	1046.1
8.	Turbine Bypass Capacity, % Rated	~~25.8%~~ 25.4%
9.	Core Coolant Inlet Enthalpy, BTU/lb (At 100% Power and 108 Mlbs/hr)	~~526.0~~(2) 526.4
10.	Turbine Inlet Pressure,psia	~~997.0~~ 997.5
11.	Fuel Types	~~SPC 9x9-2,~~ ATRIUM™-10~~, plus four GE12 LUAs~~
12.	Core Average Gap Conductance, BTU/hr-ft^2-°F	500 to 1500(1)
13.	Core Leakage Flow,%	~10%(2)
14.	Required MCPR Operating Limit	See Unit 2 COLR (FSAR section 16.3 – TRMs)
15.	MCPR Safety Limit	See Table 15D.0-3
16.	Doppler Coefficient	See Note 4

TABLE 15D.0-3

MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT (ALL FUEL)
UNIT 2 CYCLE ~~10~~ 11

MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT (ALL FUEL)
FOR
SINGLE LOOP OPERATION

~~1.11~~ 1.14

MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT (ALL FUEL)
FOR
TWO LOOP OPERATION

1.12

TABLE 15D.0-4

UNIT 2 CYCLE ~~10~~ 11

MINIMUM MCPR OPERATING LIMIT REQUIREMENT
FOR
SINGLE LOOP OPERATION

~~1.70~~ 1.89

(Based on Analysis of Pump Seizure Accident in Single Loop Operation)

MINIMUM MCPR OPERATING LIMIT REQUIREMENT
FOR
TWO LOOP OPERATION

~~1.32~~ 1.42

(Based on Analysis of Pump Seizure Accident in Two Loop Operation)

TABLE 15D.0-5

AVERAGE SCRAM INSERTION TIMES
UNIT 2 CYCLE ~~10~~ 11

Control Rod Position	Average Scram Time to Position (seconds)	
	Realistic	Maximum Allowable
45	0.470	0.520
39	0.630	0.860
25	1.500	1.910
5	2.700	3.440
Scram Time Fraction	0.0	1.0

TABLE 15D.1.2-1

SEQUENCE OF EVENTS FOR FEEDWATER CONTROLLER FAILURE, MAXIMUM DEMAND UNIT 2 CYCLE ~~10~~ 11

TIME, SECONDS	EVENT
0	Initiate simulated failure of ~~135%~~ 132.9 % upper limit on feedwater flow.
~~21.82~~ 22.35	L8 vessel level setpoint trips main turbine and feedwater pumps.
~~21.90~~ 22.42	Reactor scram trip actuated from main turbine stop valve position switch.
~~21.92~~ 22.45	Bypass Valves actuated
~~22.00~~ 22.52	Recirculation pump trip (RPT) actuated by stop valve position switch.
~~23.20~~ 23.62	First group of safety/relief valves activate due to high pressure.
~~23.32~~ 23.74	Second group of safety/relief valves activate due to high pressure.
~~23.43~~ 23.86	Third group of safety/relief valves activate due to high pressure.
~~23.61~~ 23.97	Fourth group of safety/relief valves activate due to high pressure.
-	Fifth group of safety/relief valves do not activate.

Initial Conditions:

Power	= 94%
Flow	= 108 Mlbs/hr
Bypass	= Operable
RPT	= Operable
Scram Time	= Realistic
Exposure	= EOC

TABLE 15D.2.2-1

SEQUENCE OF EVENTS FOR GENERATOR LOAD REJECTION WITHOUT BYPASS AND TURBINE TRIP WITHOUT BYPASS UNIT 2 CYCLE ~~10~~ 11

TIME, SECONDS	EVENTS
~0	Turbine-generator detection of loss electrical load.
0	Turbine-generator power load unbalance (PLU) devices trip to initiate turbine control valve fast closure.
0	Turbine bypass valves fail to operate.
0.001	Turbine control valves (TCV) close on GLR, (Generator Load Reject)
0.071	Initiate scram on TCV fast closure (Trip oil pressure-low)
0.088[1] e	Turbine control valves closed.
0.176	EOC-Reactor Pump Trip initiated.
~~0.921~~ 0.922	Group 1 relief valves actuated.
~~0.996~~ 0.998	Group 2 relief valves actuated.
~~1.080~~ 1.083	Group 3 relief valves actuated.
~~1.22~~ 1.239	Group 4 relief valves actuated.
~~1.402~~ 1.426	Group 5 relief valves actuated.

Initial Conditions

Power: ~~102%~~ 100.6 % — Flow: 108 Mlbs/hr
Bypass: Inoperable — Scram: Realistic Time
RPT: Operable

e

[1]The TCV closure time corresponding to 100% rated power was used to be conservative.

TABLE 15D.3.3-1

PUMP SEIZURE ACCIDENT FROM TWO LOOP OPERATION SEQUENCE OF EVENTS UNIT 2 CYCLE ~~10~~ 11

TIME, SEC	EVENT
0.0	Single Pump Seizure was Initiated
~~0.7~~ 0.6	Jet Pump Diffuser Flow Reverses in Seized Loop
~~1.3~~ 1.10	Vessel Level (L8) Trip
~~1.4~~ 1.29	Minimum CPR
~~2.3~~ 2.15	Vessel Level (L8) Trip Initiates Turbine Stop Valve Motion
~~2.4~~ 2.22	Turbine Stop Valve Motion Initiates Reactor Scram

TABLE 15D.3.3-2

PUMP SEIZURE ACCIDENT FROM SINGLE LOOP OPERATION
SEQUENCE OF EVENTS
UNIT 2 CYCLE ~~10~~ 11

TIME, SEC	EVENT
0.0	Single Pump Seizure was Initiated
N/A	Jet Pump Diffuser Flow Reverses in Seized Loop
~~1.3~~ 1.19	Vessel Level (L8) Trip
~~1.9~~ 2.03	Minimum CPR
~~2.3~~ 2.25	Vessel Level (L8) Trip Initiates Turbine Stop Valve Motion
~~2.4~~ 2.33	Turbine Stop Valve Motion Initiates Reactor Scram

TABLE 15D.3.3-7

DOSES FOR THE SINGLE LOOP AND TWO LOOP PUMP SEIZURE EVENT
UNIT 2 CYCLE ~~10~~ 11

		Dose at the Site Boundary(1) (REM)	
		ATRIUM™-10	Limit
Two Loop Operation (~~7681~~ 7741 failed rods)	Thyroid	~~7.47~~ 7.53	30
	Whole Body	~~1.28~~ 1.29	2.5
Single Loop Operation (~~7621~~ 7986 failed rods)	Thyroid	~~7.42~~ 7.77	30
	Whole Body	~~1.27~~ 1.33	2.5

		Dose at Low Population Zone(1) (Rem)	
		ATRIUM™-10	Limit
Two Loop Operation (~~7681~~ 7741 failed rods)	Thyroid	~~3.88~~ 3.91	30
	Whole Body	~~0.66~~ 0.67	2.5
Single Loop Operation (~~7621~~ 7986 failed rods)	Thyroid	~~3.85~~ 4.03	30
	Whole Body	~~0.66~~ 0.69	2.5

Note 1: All failed rods are assumed to be full length ATRIUM™-10 rods. ~~For cycle 10, the majority of the fuel assemblies are the ATRIUM™-10 design (592 of 764 assemblies).~~

TABLE 15D.3.3-8

INPUT PARAMETERS AND INITIAL CONDITIONS FOR PUMP SEIZURE ACCIDENT
(SINGLE LOOP AND TWO LOOP OPERATION)
UNIT 2 CYCLE ~~10~~ 11

	Design Basis Assumptions
I. Data and Assumptions Used to Estimate Radioactive Source	
A. Reactor Power (MWt)	3616
B. Fuel Damaged due to dryout (Clad Damage) 1. Number of failed fuel rods in two loop operation 2. Number of failed fuel rods in single loop operation	 ~~7681~~ 7741 ~~7021~~ 7986
C. Number of Fuel Rods per Bundle 1. ATRIUM™-10 Fuel Design	 87.8[1]
D. Number of Fuel Bundles in Core	764
E. Radial Power Peaking Factor	1.5
F. Decay Time After Full Power Operation Assumed for Fuel Rod Activity (minutes)	0
G. Release of Activity by Nuclide from Failed Fuel 1. ATRIUM™-10	 Table 15D.3.3-3
H. Iodine Fractions 1. Organic 2. Elemental 3. Particulate	 0 1.0 0
I. Reactor coolant Activity Before the Accident	0

TABLE 15D.4.5-1

SEQUENCE OF EVENTS FOR RECIRCULATION FLOW CONTROLLER FAILURE UNIT 2 CYCLE ~~10~~ 11

TIME, SECONDS	EVENT
0	Simulate failure of Master Flow Controller at ~~0.2374~~%/sec. 0.2700 %
~~224.4~~ 201.0	Reactor high pressure scram (analytical setpoint, 1119.7 psia).
~~225.2~~ 200.9	Two relief valves open at 1120.7 psia.
~~228.1~~ 204.5	Two relief valves reseat at 1045.7 psia.

This sequence of events is for the event initiated from:

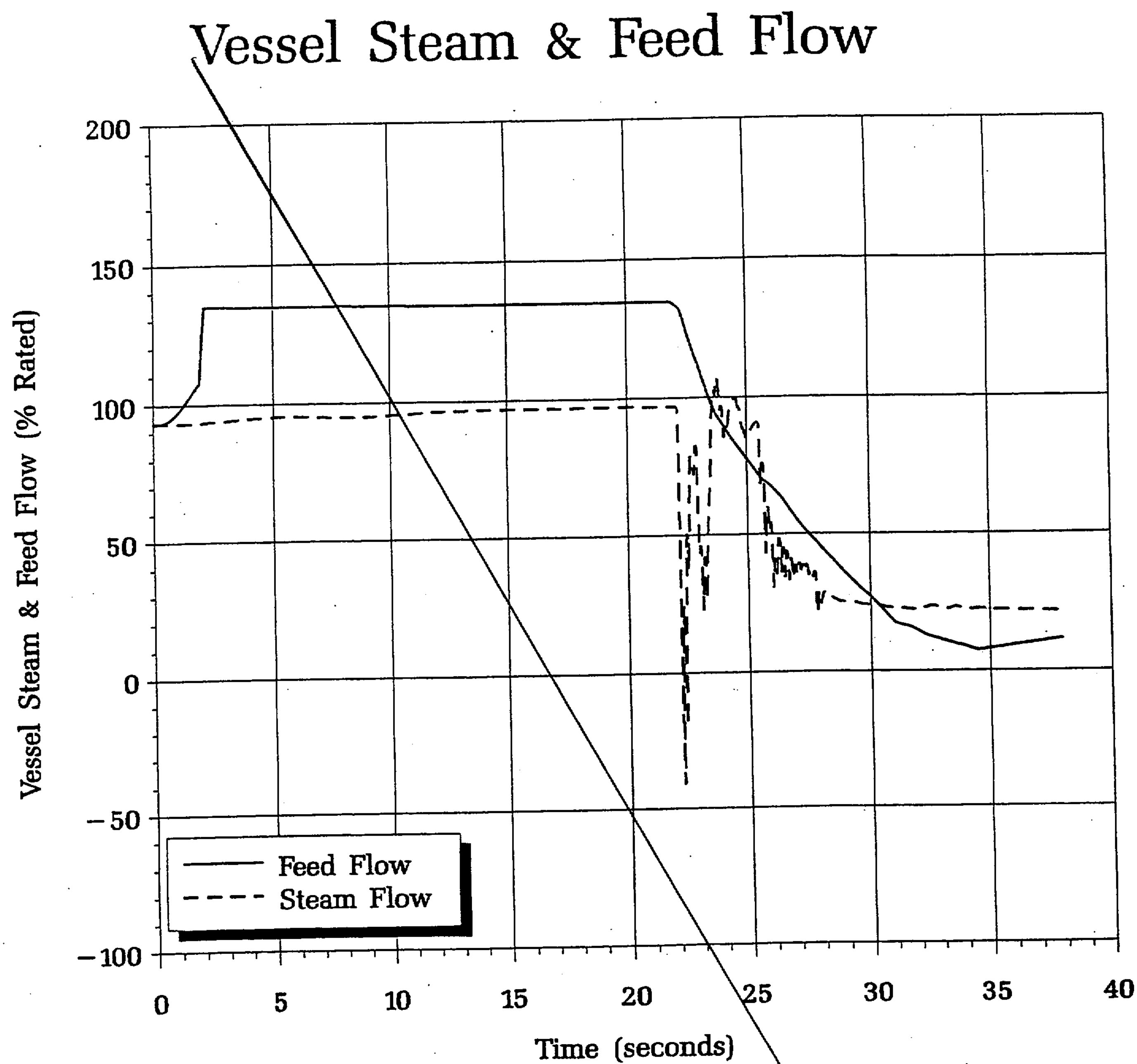
68% ~~69~~% Power

60 Mlbs/hr Flow

TABLE 15D.4.9-2

CONTROL ROD DROP ACCIDENT
UNIT 2 CYCLE ~~10~~ 11

Cycle Exposure, MWD/MTU	~~17.004~~ 17,474
Control Rod Sequence	~~B2~~ B1
Rod Group	~~10~~ 8
Dropped Rod Location	~~06-35~~ 26-55
Dropped Rod Worth	12.677 ~~13.02~~ mk (from 00 to 12)
Number of Fuel Rods with Fuel Enthalpy above 170 cal/gm	637 ~~910~~
Peak deposited Enthalpy, cal/gm	226 ~~268~~



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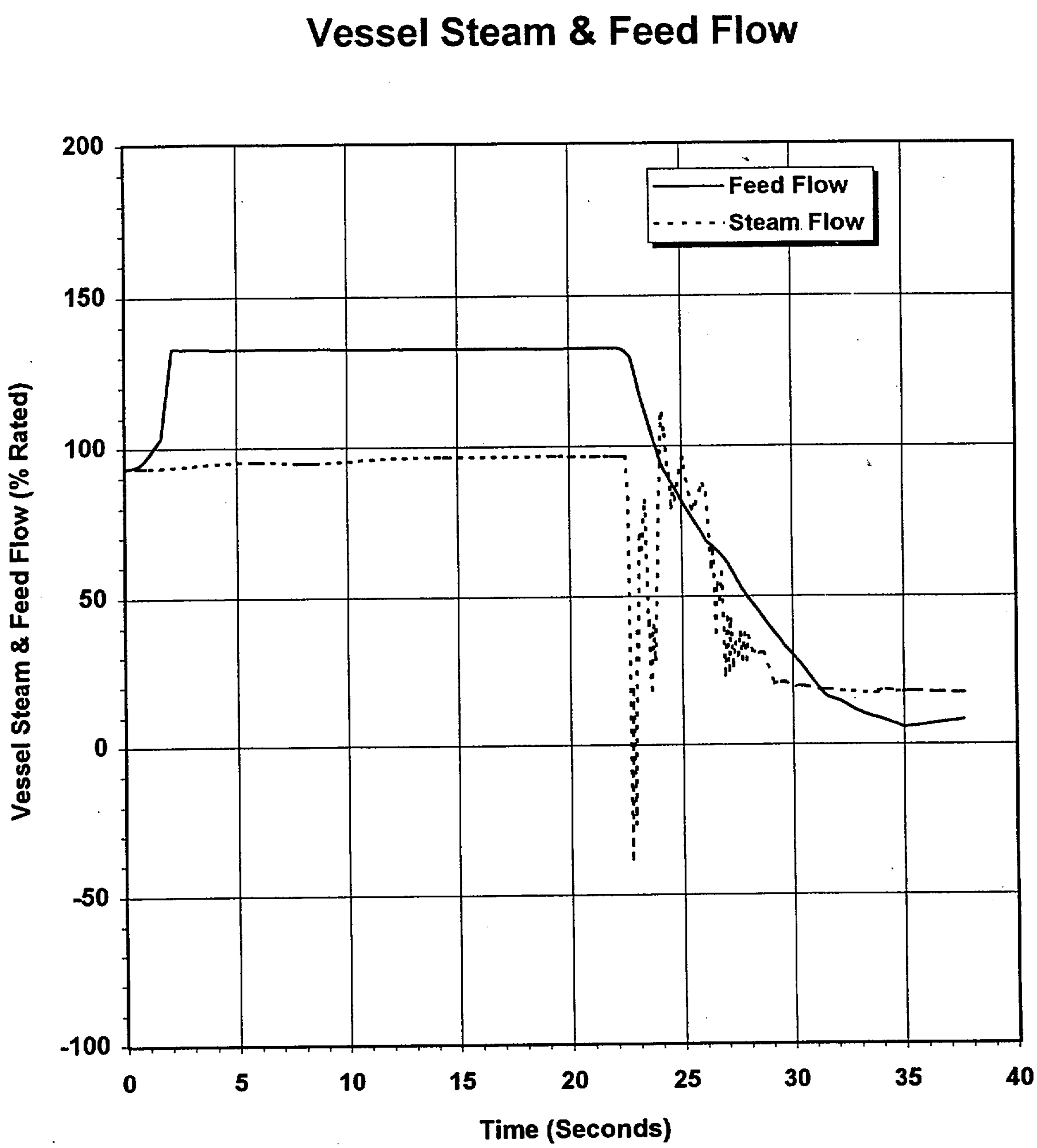
UNIT 2 CYCLE 10
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER
FAILURE, MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FSAR FIGURE 15D.1.2–1–1

NUCLEAR FUELS



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UNIT 2 CYCLE 11
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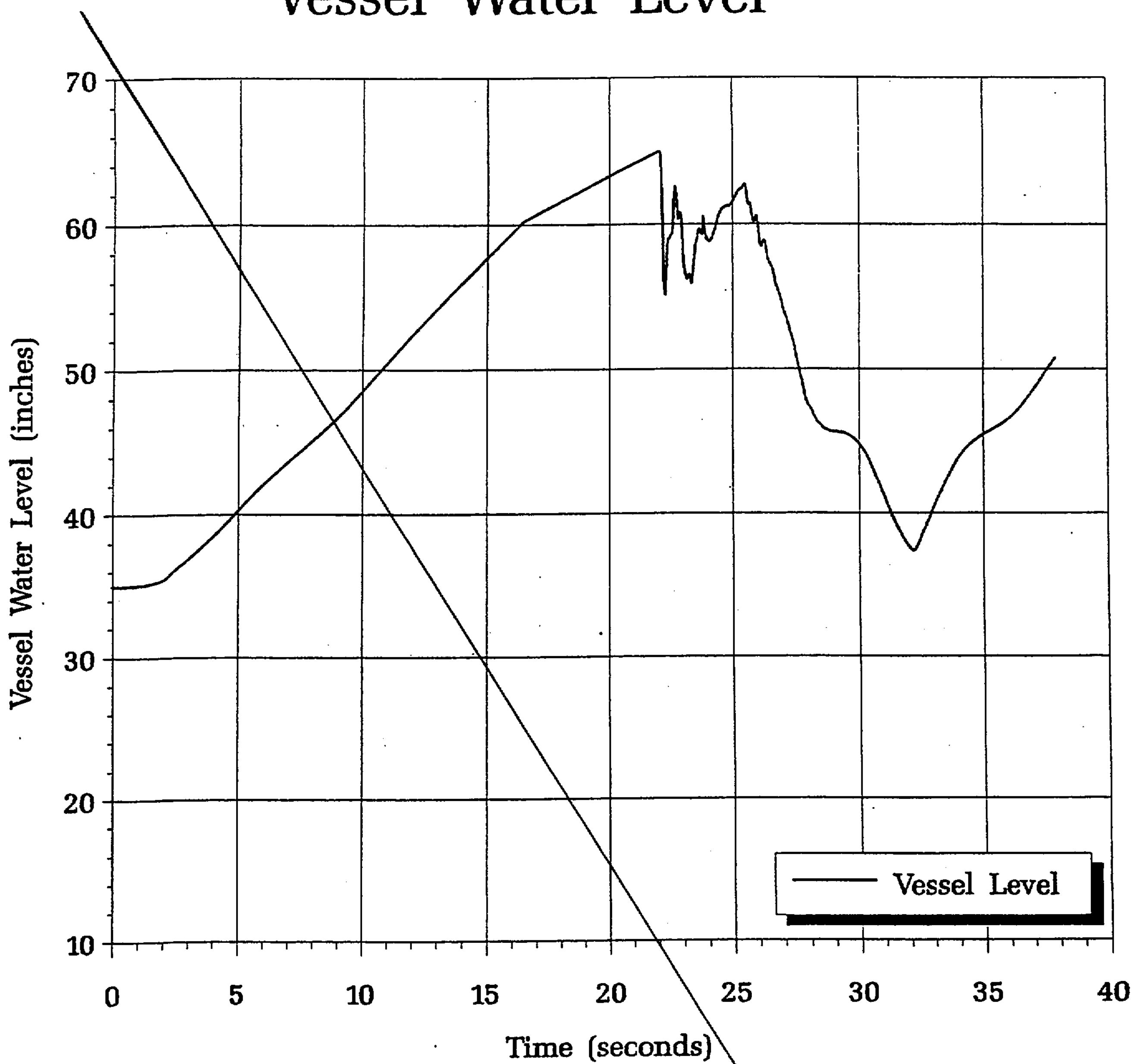
SUSQUEHANNA FEEDWATER CONTROLLER
FAILURE MAXIUMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FSAR FIGURE 15D.1.2-1-1

NUCLEAR FUELS

Vessel Water Level



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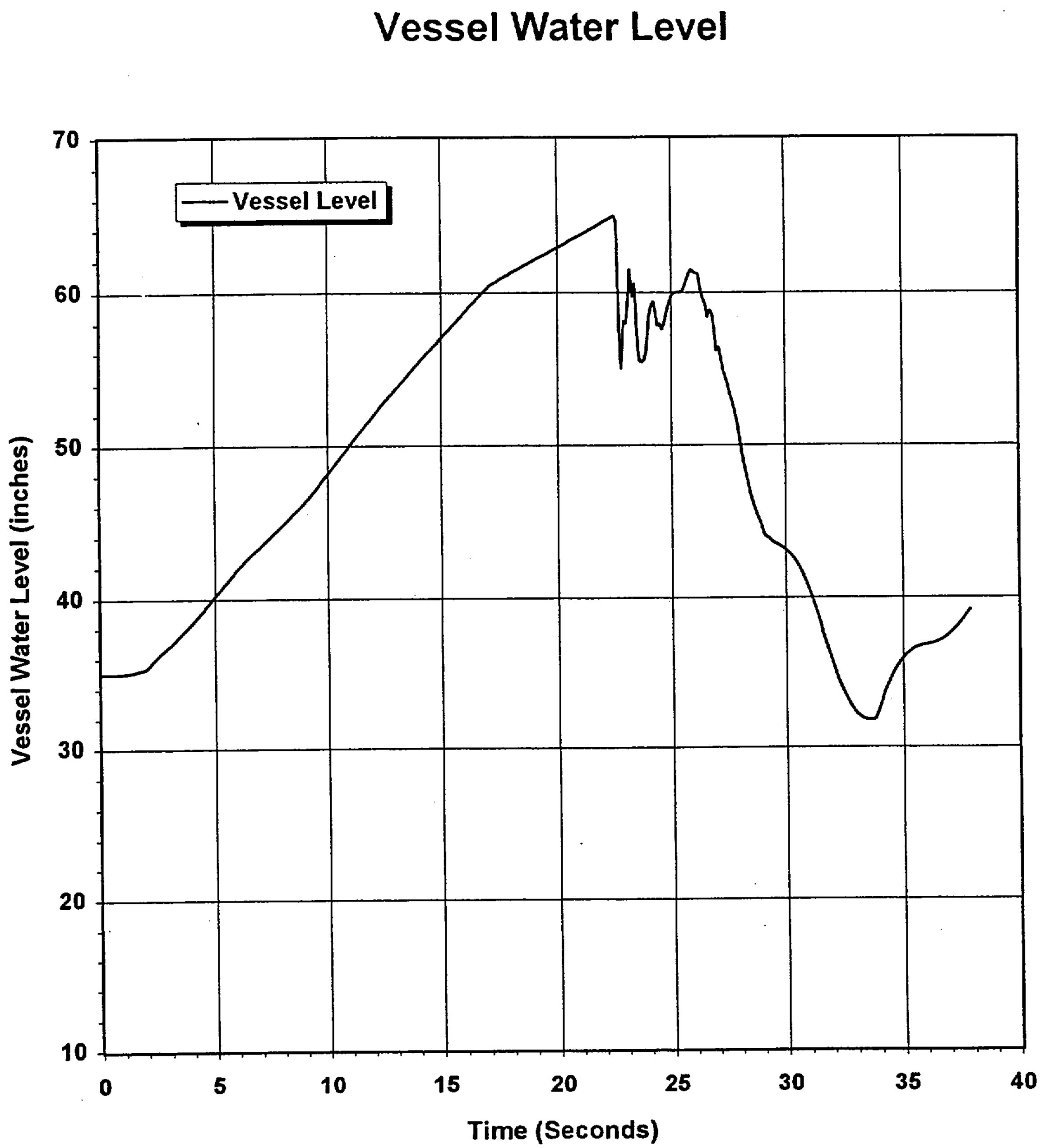
SUSQUEHANNA FEEDWATER CONTROLLER
FAILURE, MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FSAR FIGURE 15D.1.2-1-2

NUCLEAR FUELS



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UNIT 2 CYCLE 11
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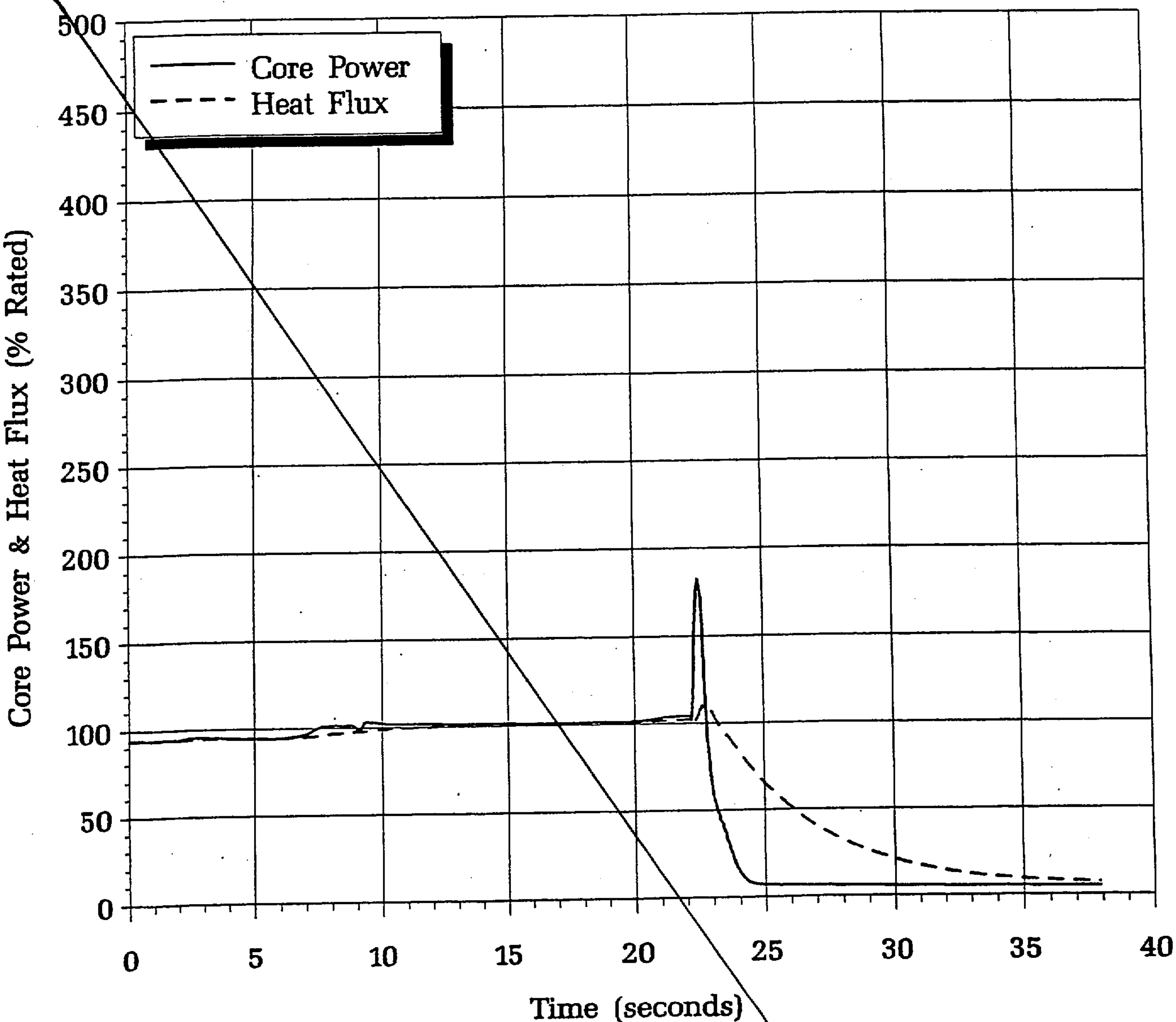
SUSQUEHANNA FEEDWATER CONTROLLER
FAILURE MAXIUMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FSAR FIGURE 15D.1.2-1-2

NUCLEAR FUELS

Core Power & Heat Flux



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UNIT 2 CYCLE 10
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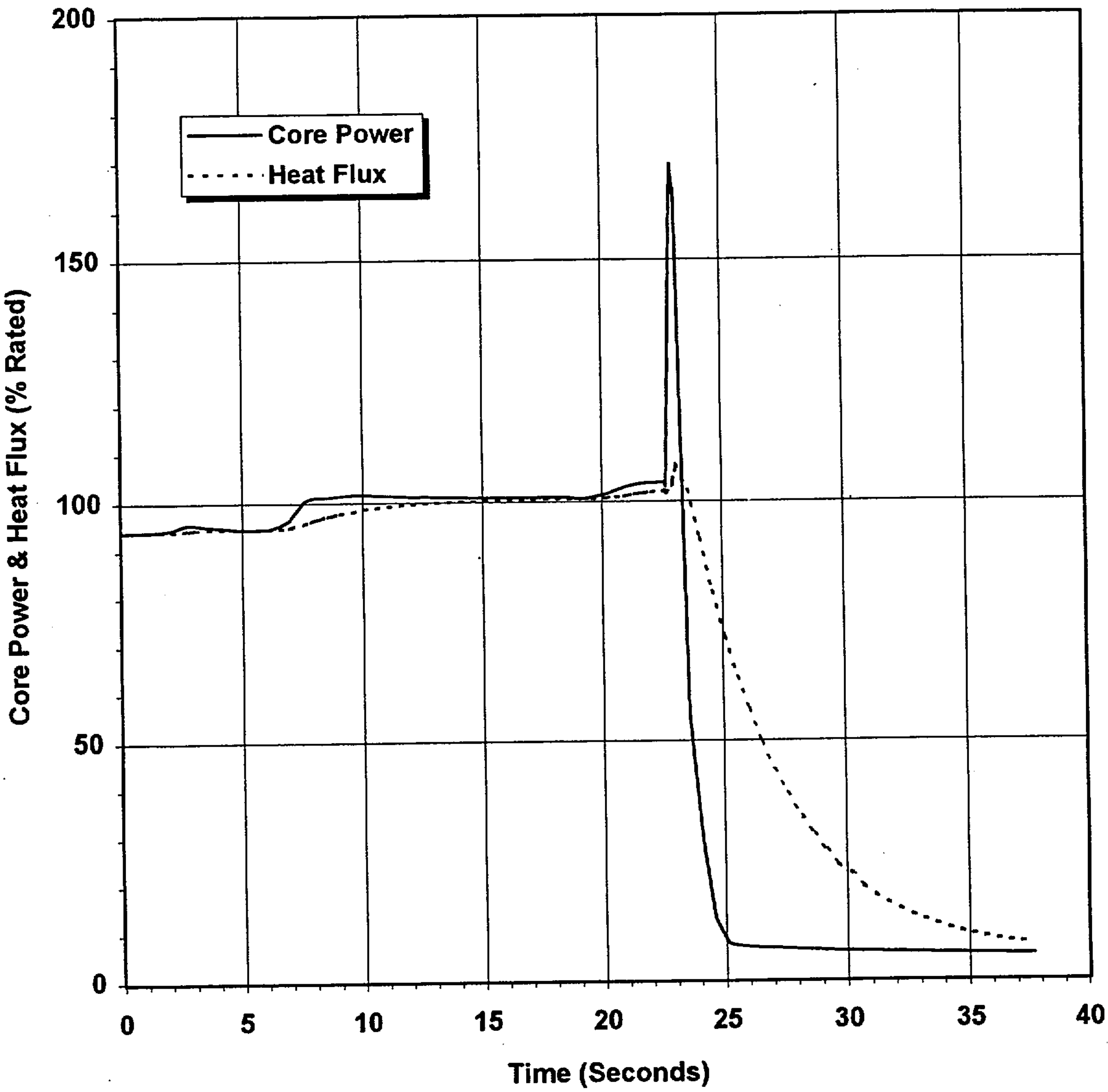
SUSQUEHANNA FEEDWATER CONTROLLER
FAILURE, MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FSAR FIGURE 15D.1.2-1-3

NUCLEAR FUELS

Core Power & Heat Flux



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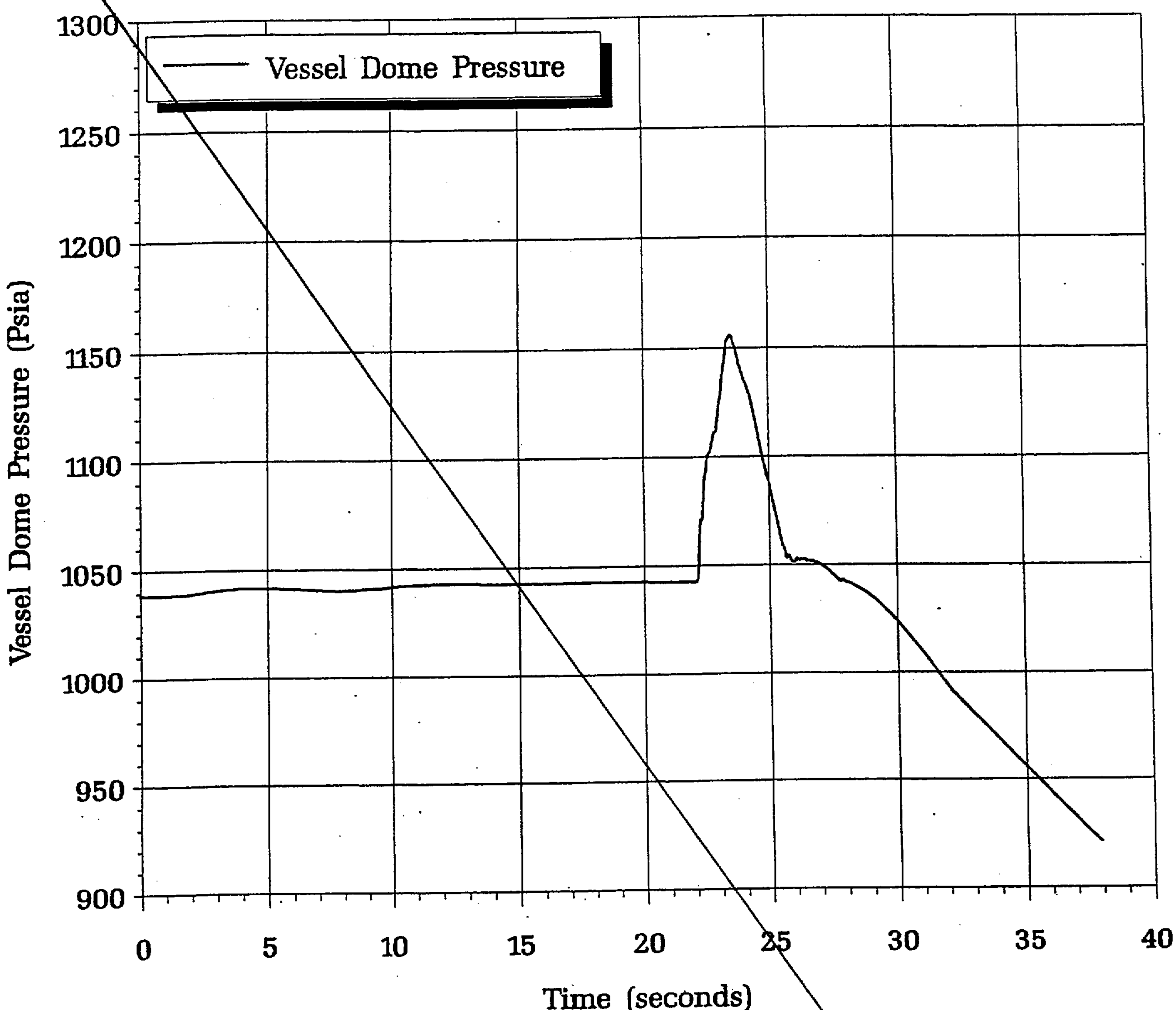
SUSQUEHANNA FEEDWATER CONTROLLER FAILURE MAXIUMUM DEMAND, WITH HIGH WATER LEVEL TRIP FSAR FIGURE 15D.1.2-1-3

NUCLEAR FUELS

Vessel Dome Pressure



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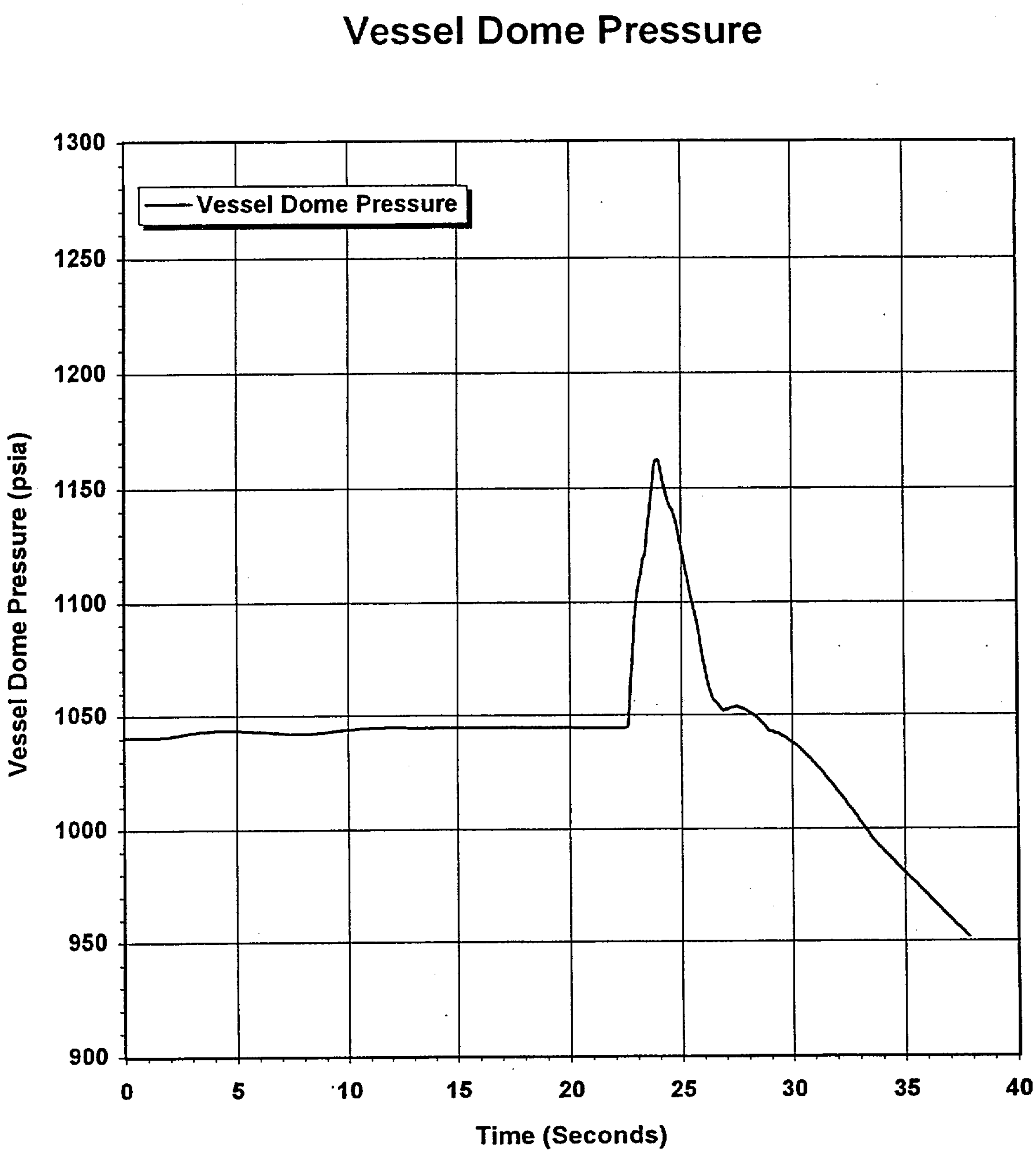
UNIT 2 CYCLE 10
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER
FAILURE, MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FSAR FIGURE 15D.1.2-1-4

NUCLEAR FUELS



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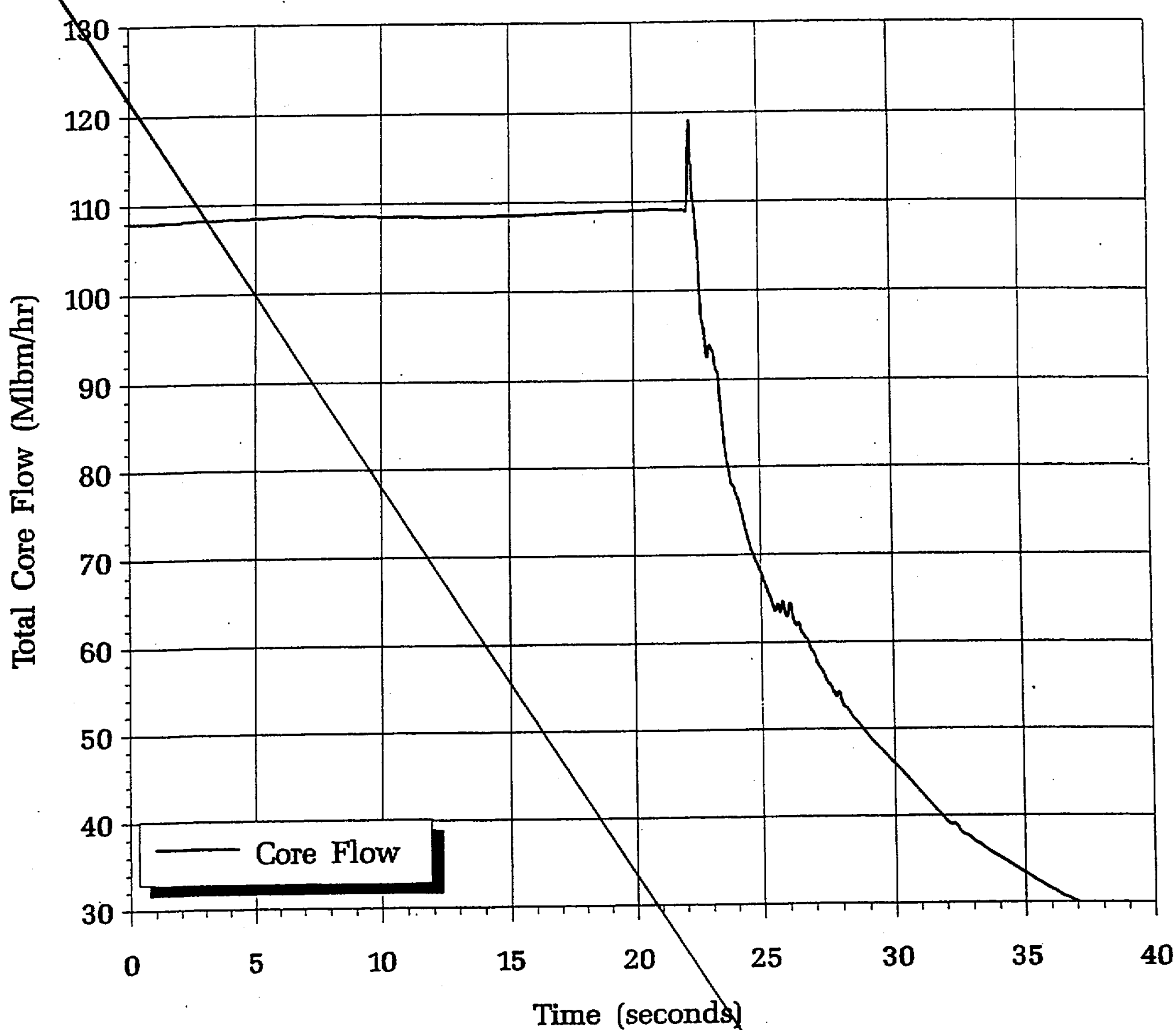
UNIT 2 CYCLE 11
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER FAILURE MAXIUMUM DEMAND, WITH HIGH WATER LEVEL TRIP

FSAR FIGURE 15D.1.2-1-4

NUCLEAR FUELS

Total Core Flow



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UNIT 2 CYCLE 10
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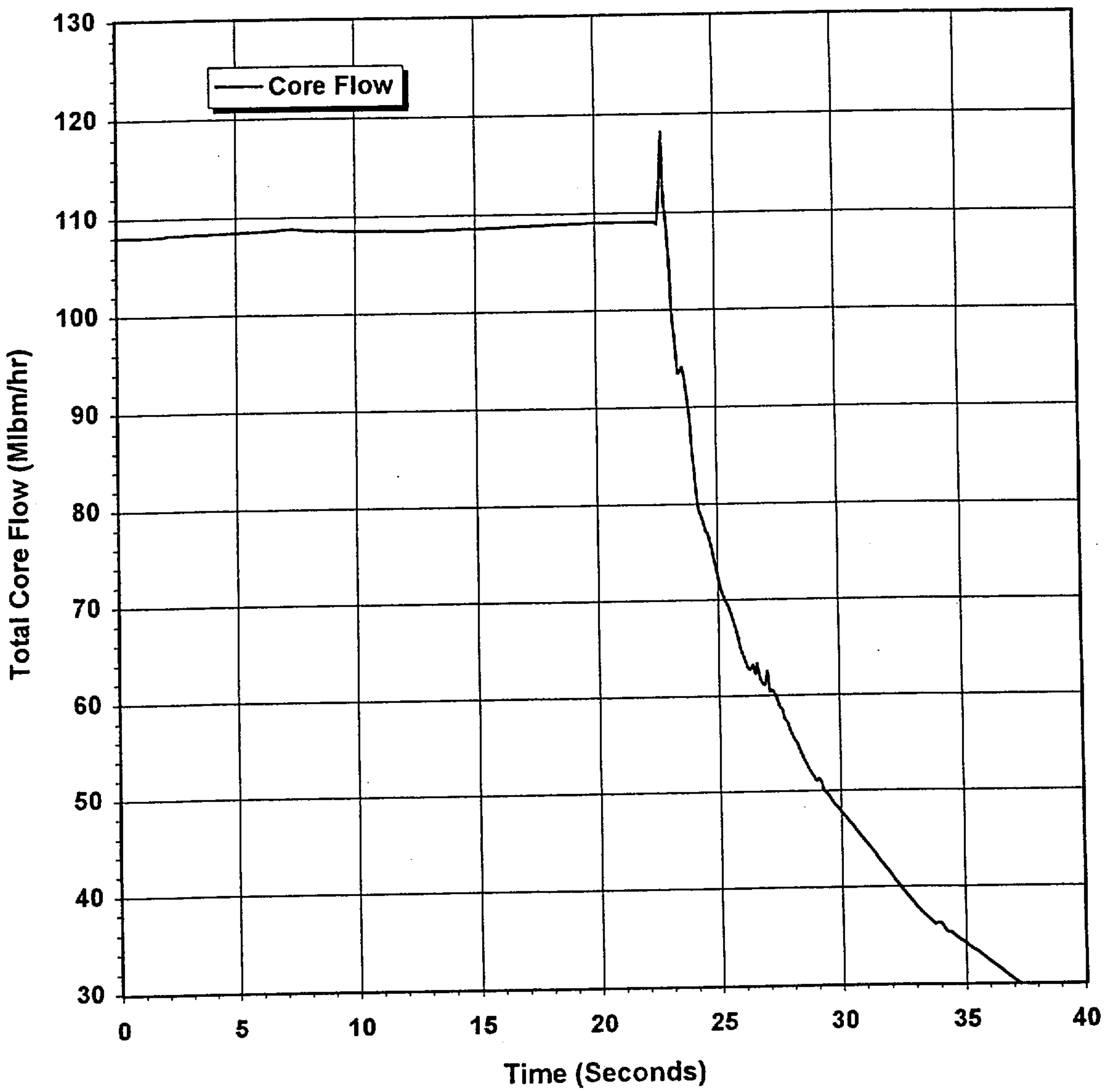
SUSQUEHANNA FEEDWATER CONTROLLER
FAILURE, MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FSAR FIGURE 15D.1.2-1-5

NUCLEAR FUELS

Total Core Flow



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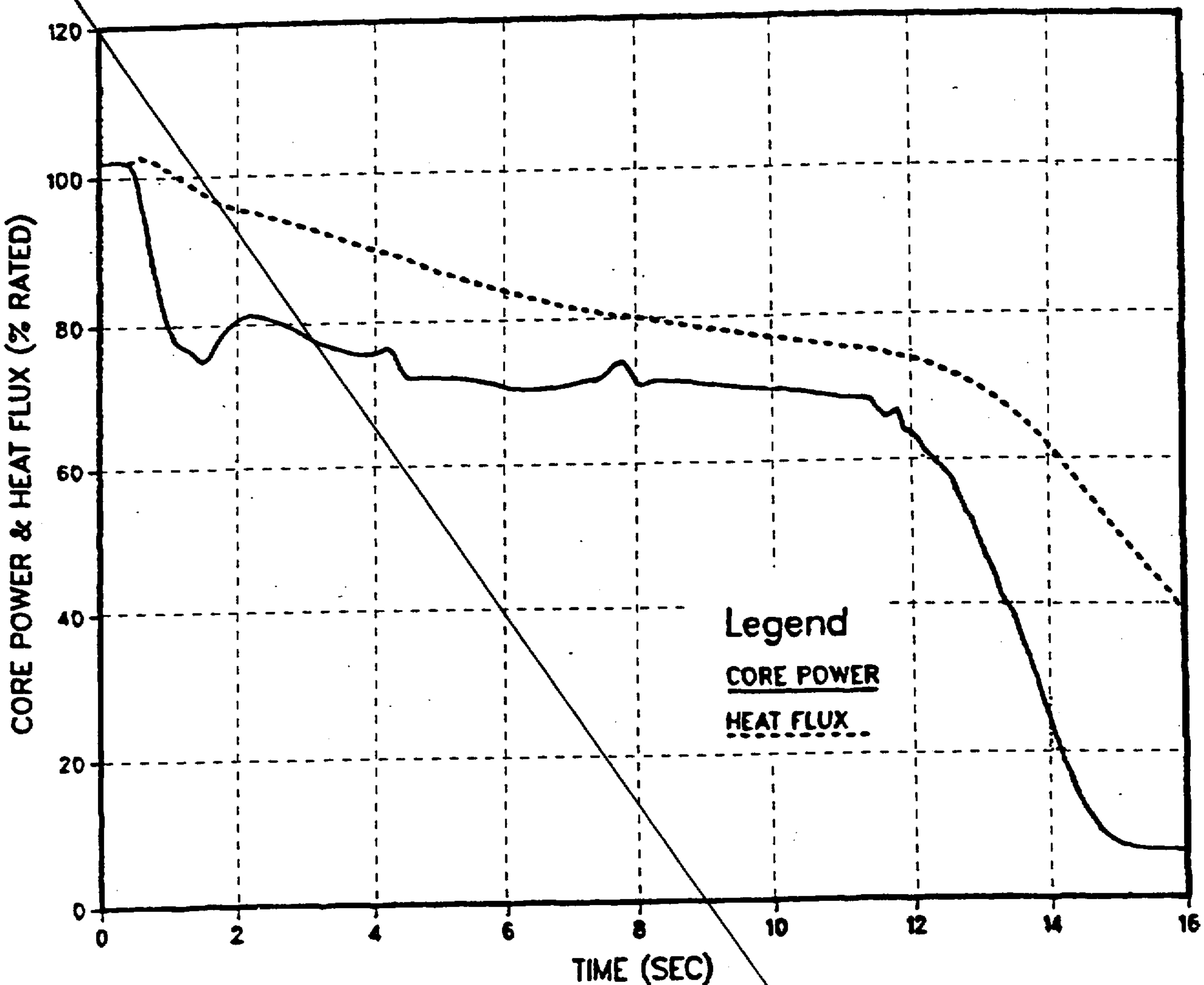
UNIT 2 CYCLE 11
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER
FAILURE MAXIUMUM DEMAND,
WITH HIGH WATER LEVEL TRIP

FSAR FIGURE 15D.1.2-1-5

NUCLEAR FUELS



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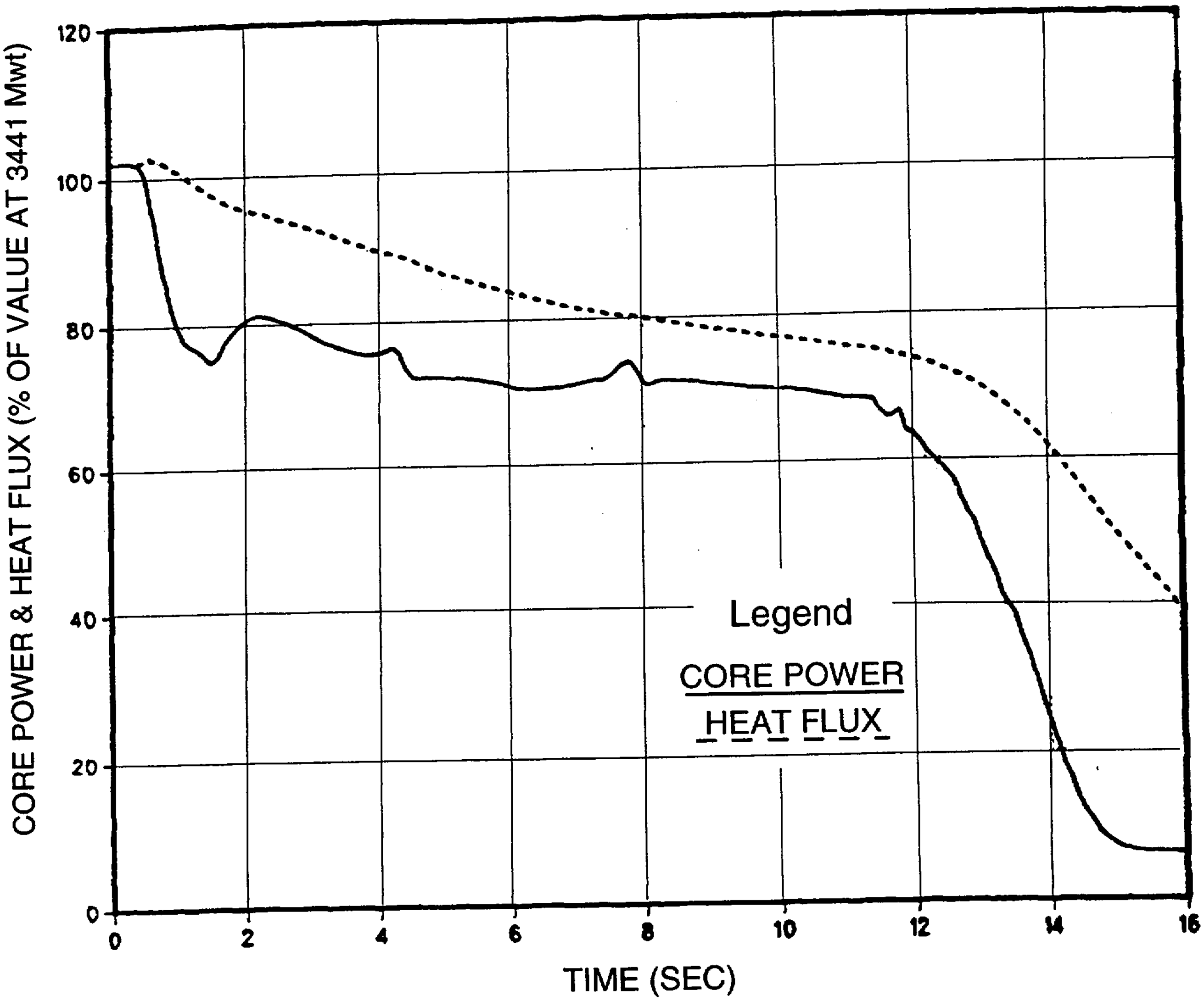
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

PRESSURE REGULATOR FAILURE - STEAM FLOW AT 130% OF RATED

FIGURE 15D.1.3-1-1

PPL DRAWING IMAGE CENTER



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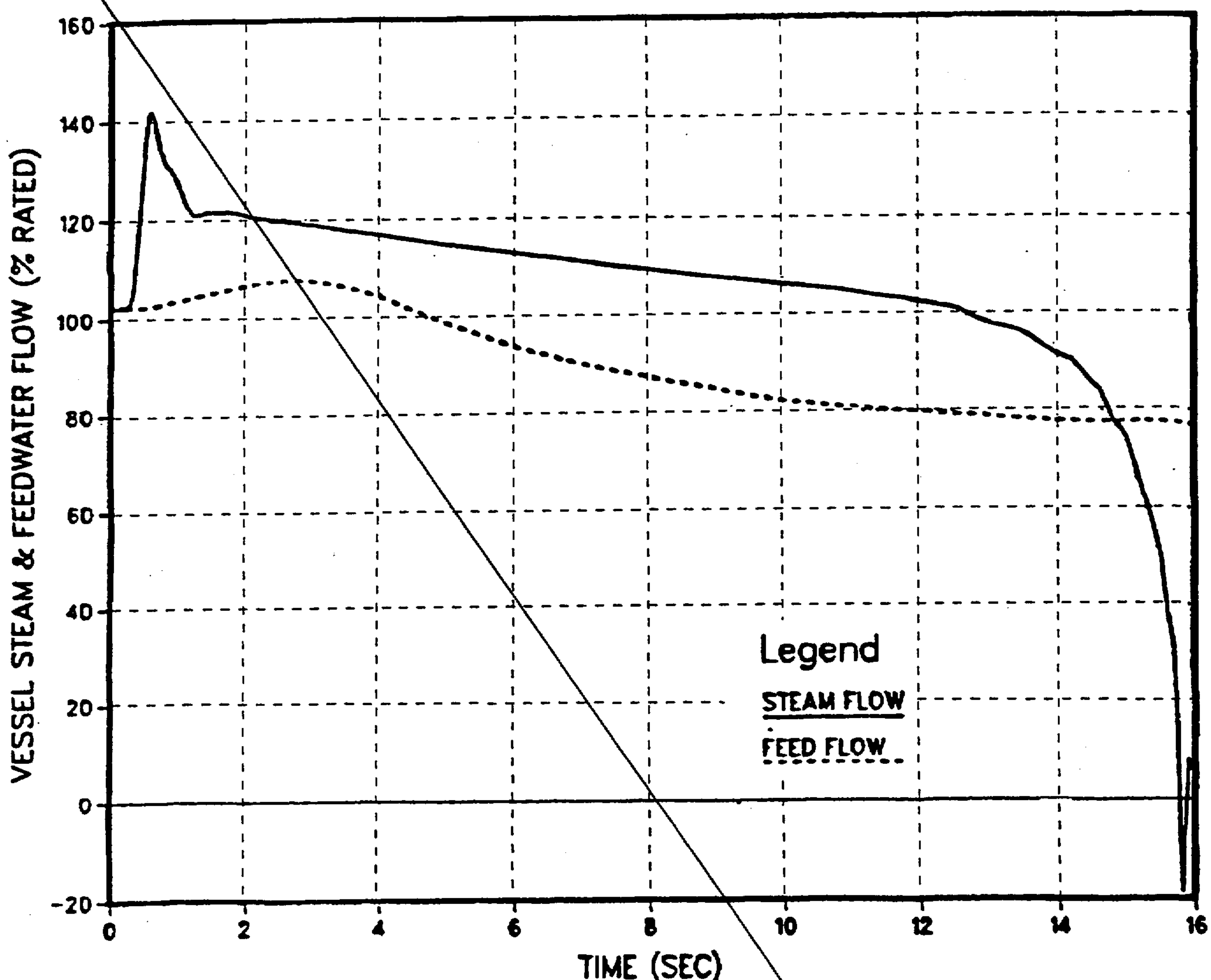
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

PRESSURE REGULATOR FAILURE - STEAM FLOW AT 130% OF RATED

FIGURE 15D.1.3-1-1, Rev. 54

PPL DRAWING IMAGE CENTER



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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

PRESSURE REGULATOR FAILURE - STEAM FLOW AT 130% OF RATED

FIGURE 15D.1.3-1-3

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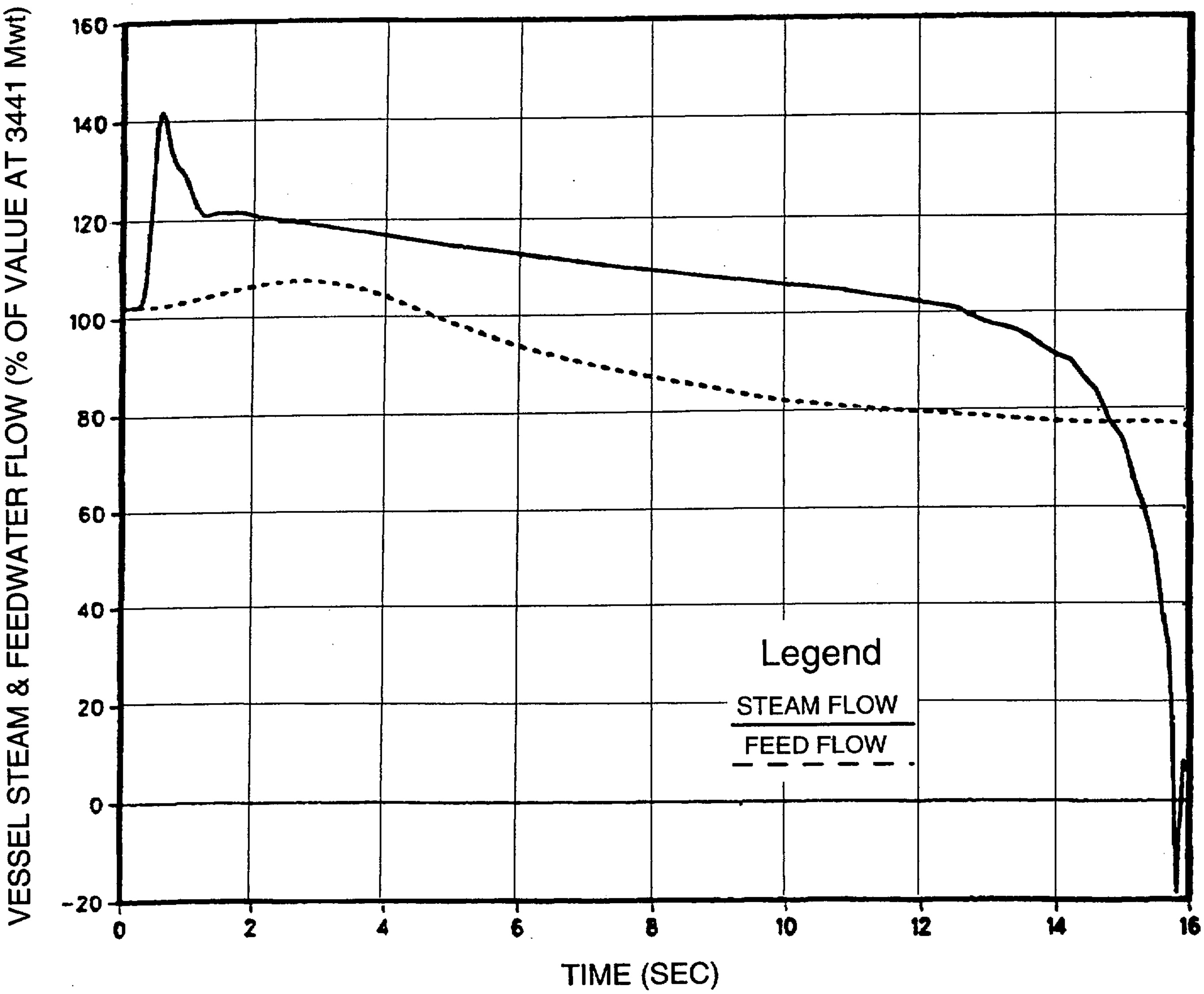

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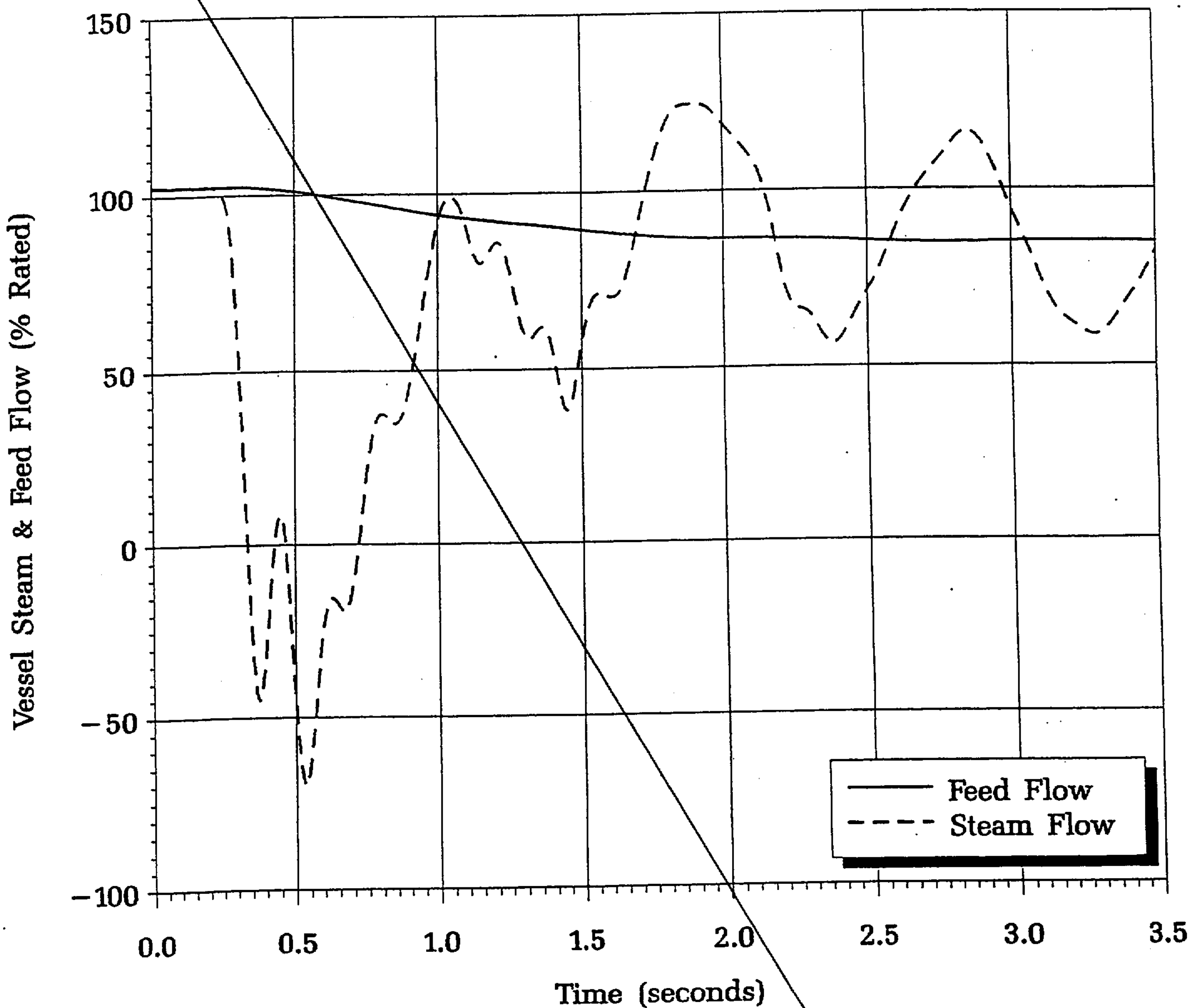
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

PRESSURE REGULATOR FAILURE -
STEAM FLOW AT 130% OF RATED

FIGURE 15D.1.3-1-3, Rev. 54

PPL DRAWING IMAGE CENTER

Vessel Steam & Feed Flow



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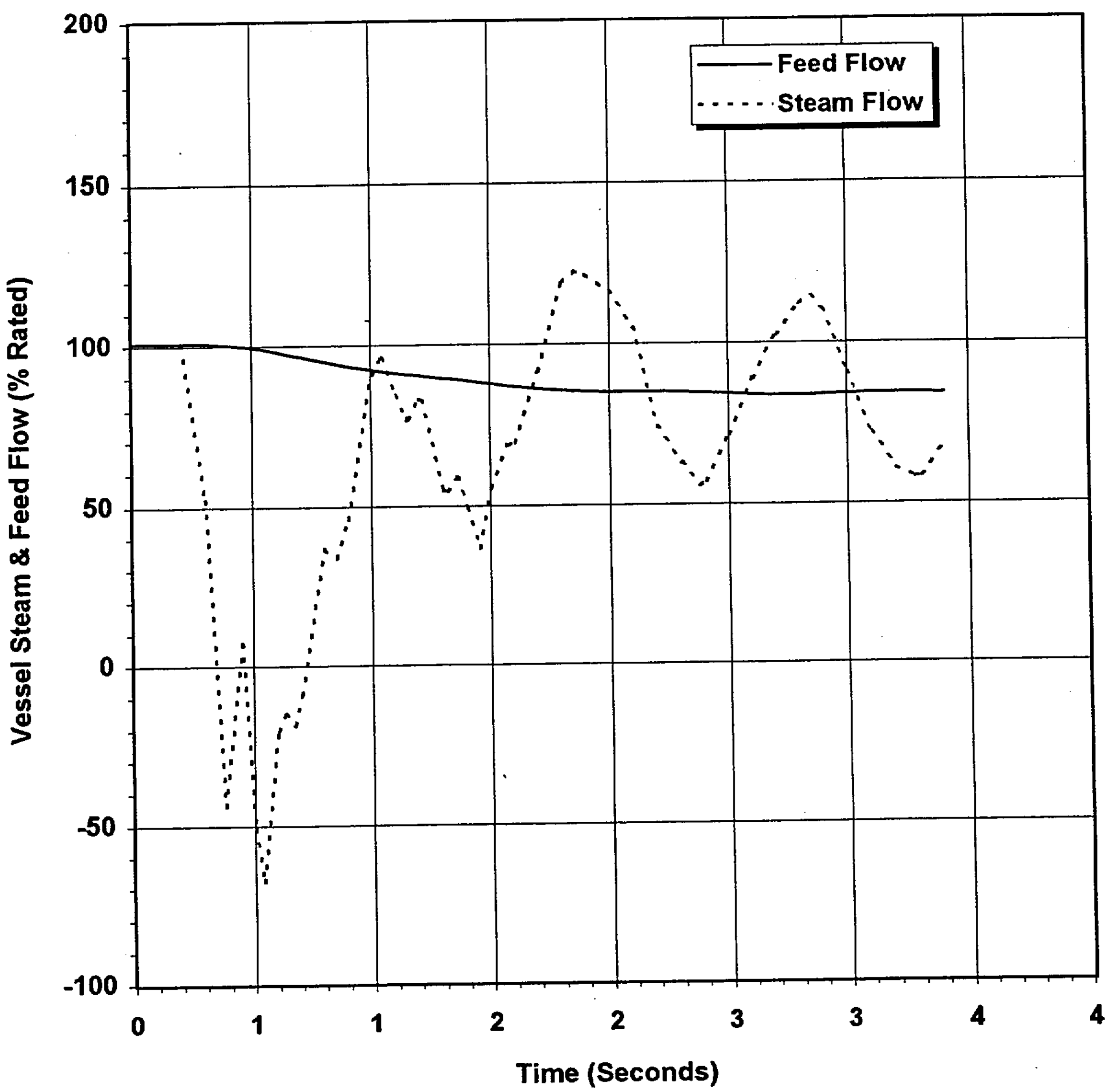
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 10
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FSAR FIGURE 15D.2.2-1-1

NUCLEAR FUELS

Vessel Steam & Feed Flow



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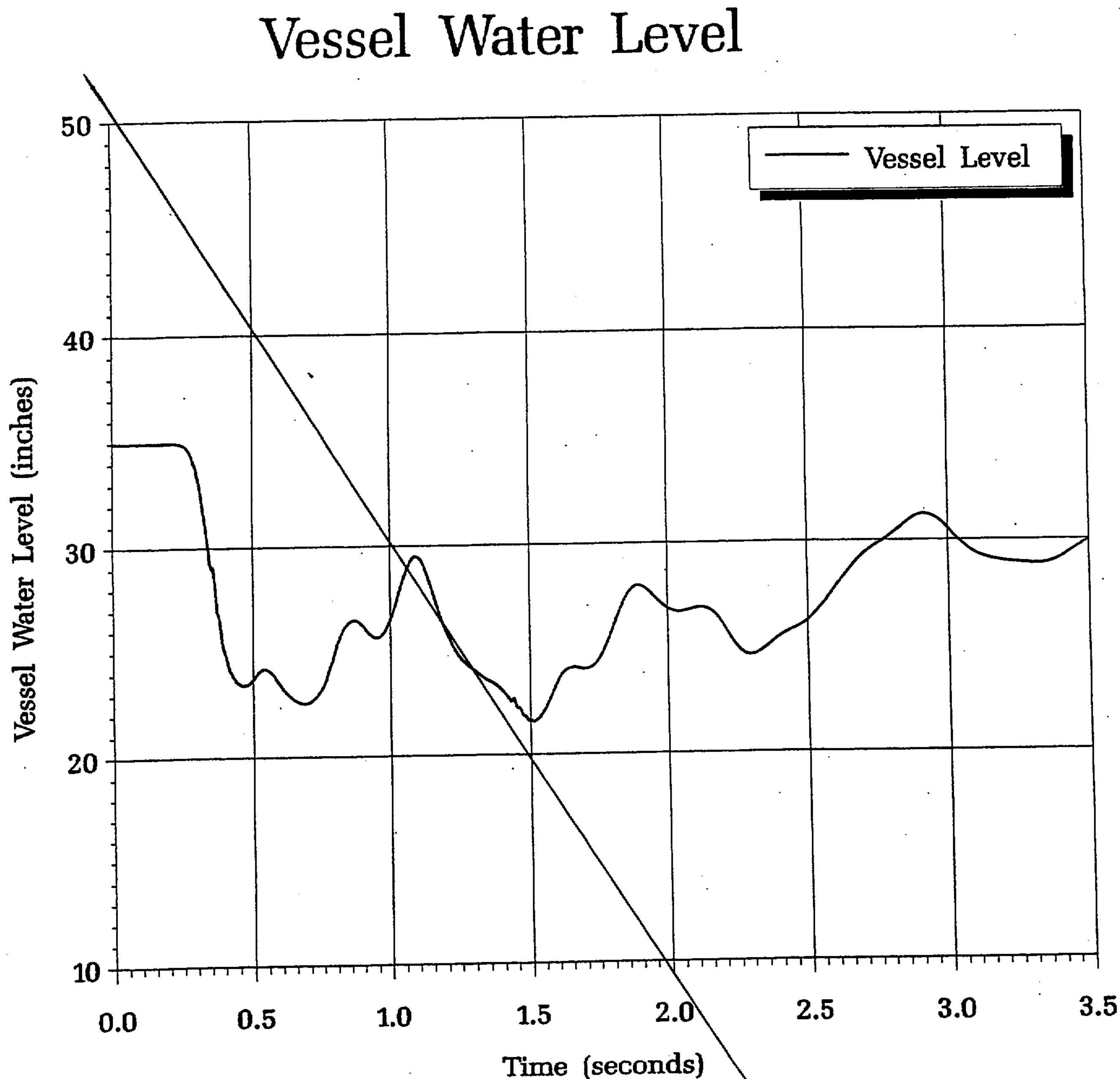
SUSQUEHANNA STEAM ELECTRIC STATION UNIT 2 CYCLE 11 FINAL SAFETY ANALYSIS REPORT
SUSQUEHANNA GENERATOR LOAD REJECT WITHOUT BYPASS AND TURBINE TRIP WITHOUT BYPASS FSAR FIGURE 15D.2.2-1-1

NUCLEAR FUELS



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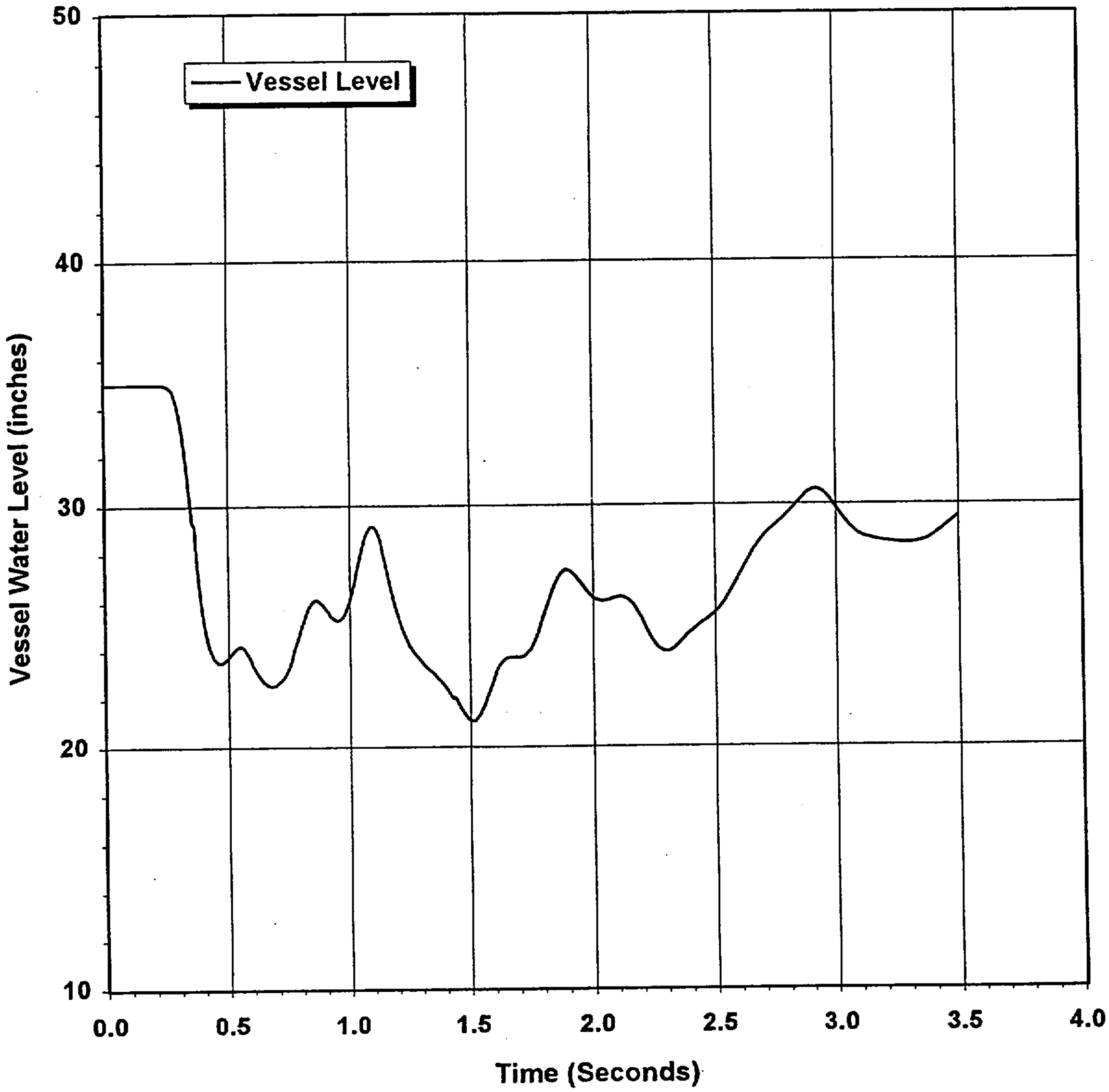
SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FSAR FIGURE 15D.2.2-1-2

NUCLEAR FUELS

Vessel Water Level



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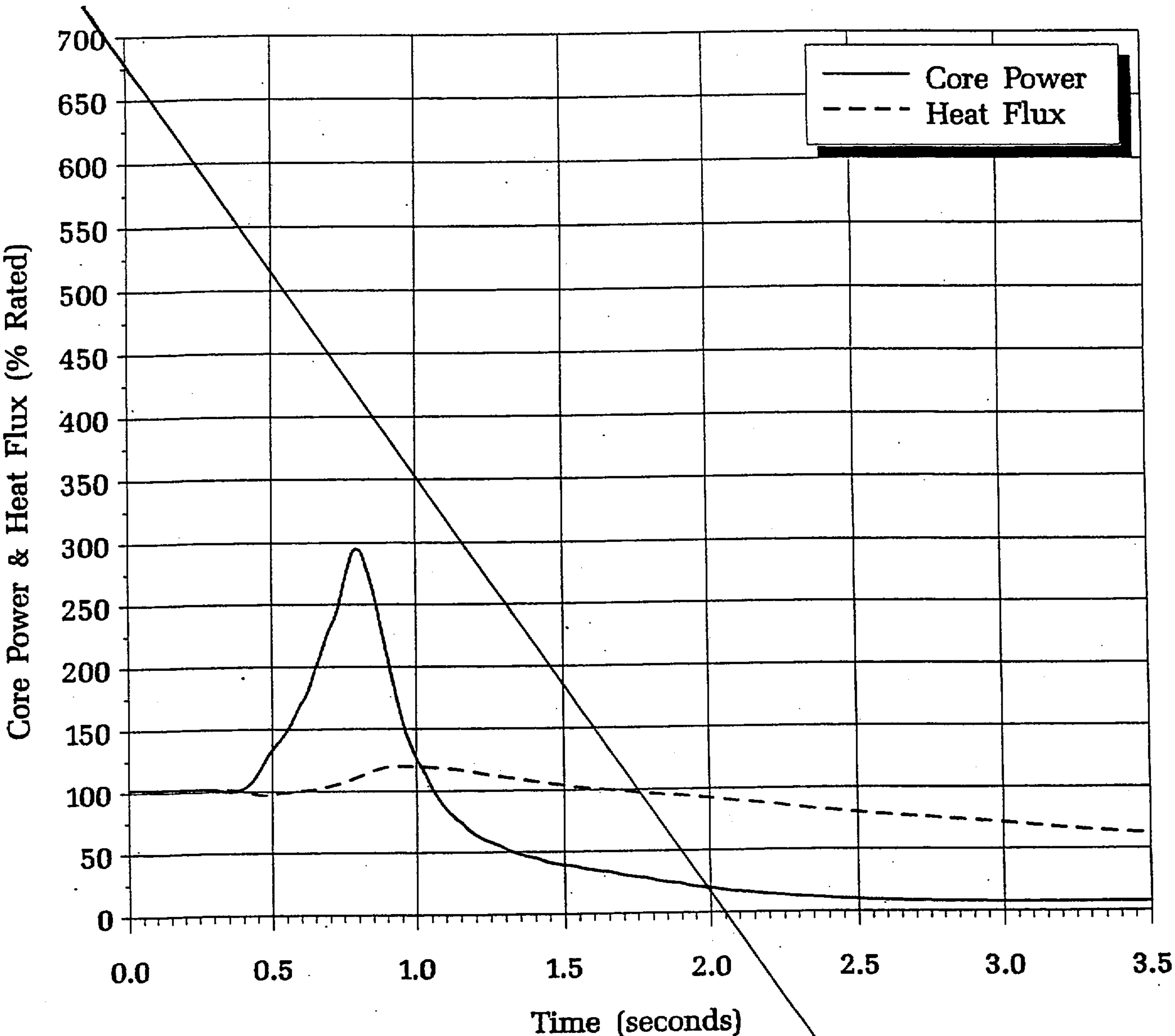
SUSQUEHANNA STEAM ELECTRIC STATION UNIT 2 CYCLE 11 FINAL SAFETY ANALYSIS REPORT
SUSQUEHANNA GENERATOR LOAD REJECT WITHOUT BYPASS AND TURBINE TRIP WITHOUT BYPASS FSAR FIGURE 15D.2.2-1-2

NUCLEAR FUELS

Core Power & Heat Flux



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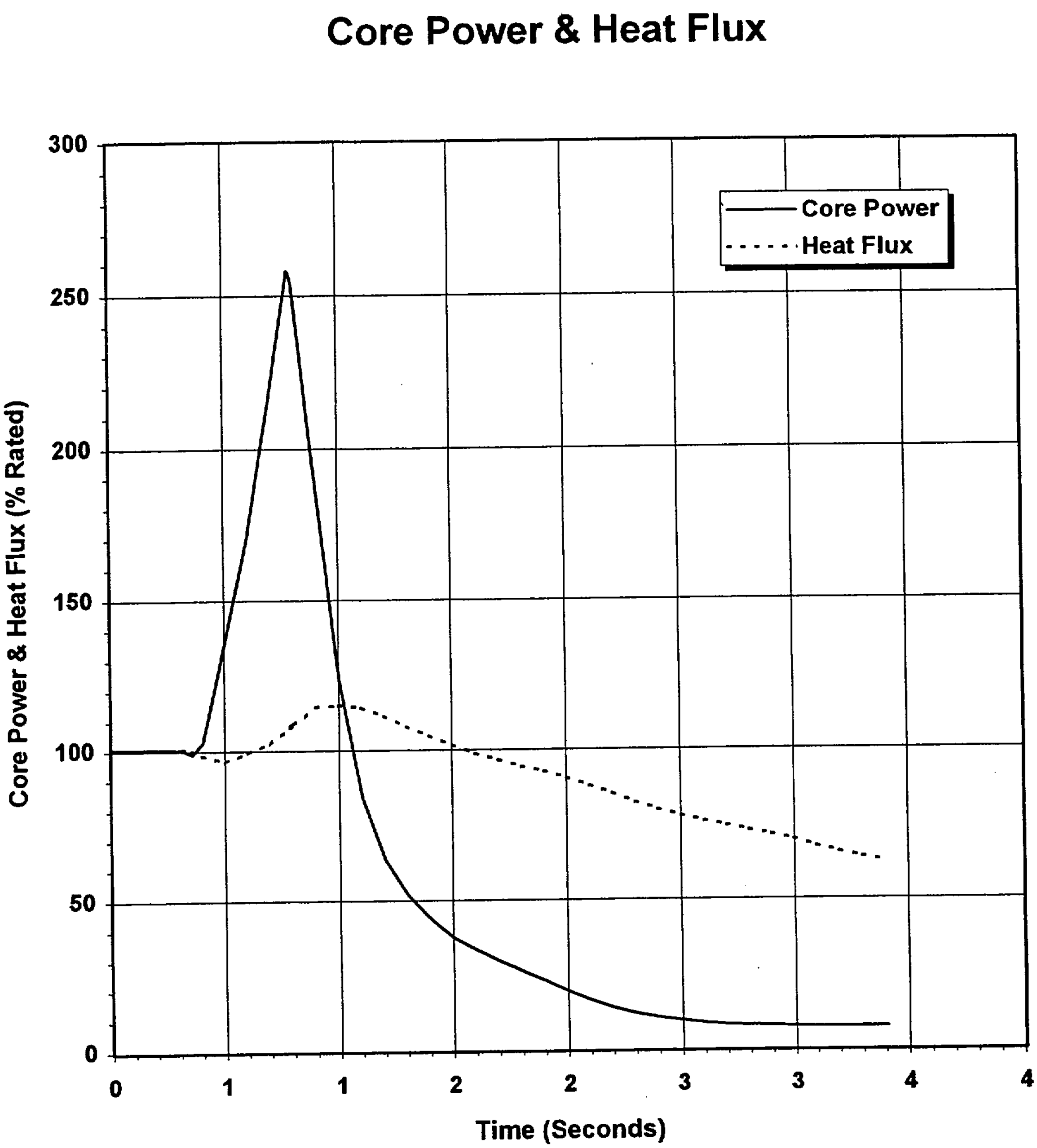
UNIT 2 CYCLE 10
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FSAR FIGURE 15D.2.2-1-3

NUCLEAR FUELS



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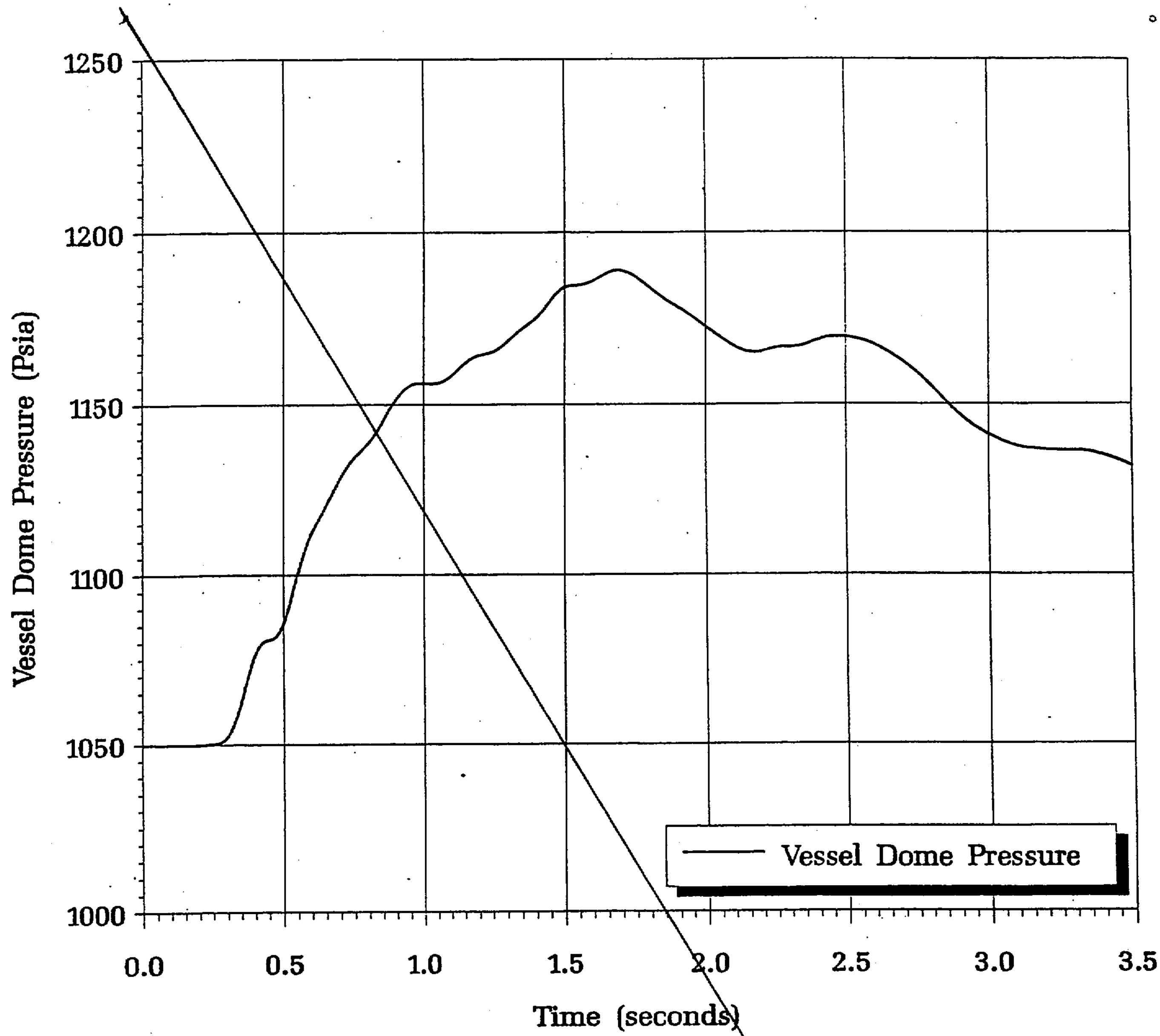
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 11
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FSAR FIGURE 15D.2.2-1-3

NUCLEAR FUELS

Vessel Dome Pressure



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SUSQUEHANNA STEAM ELECTRIC STATION
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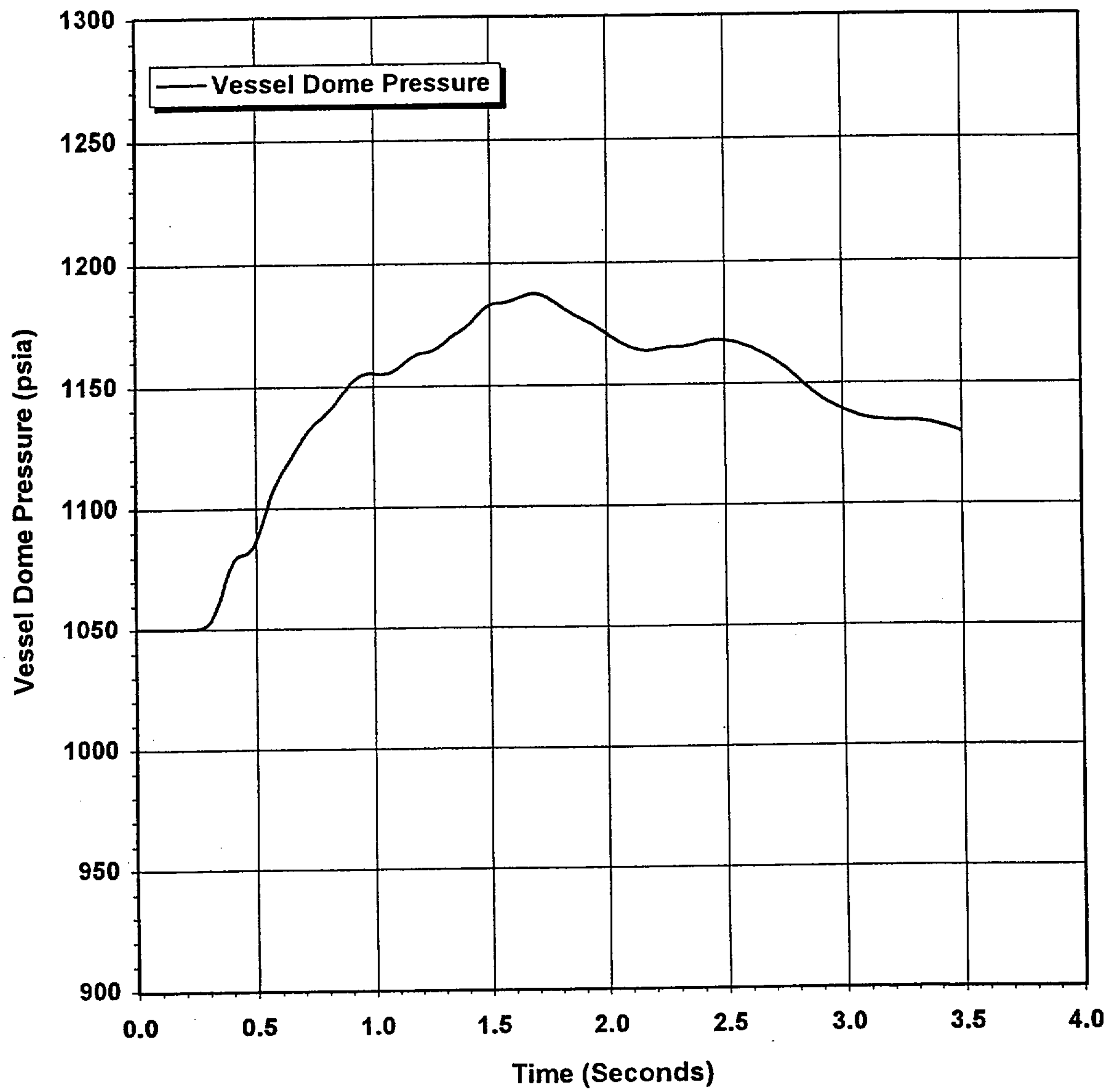
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FSAR FIGURE 15D.2.2-1-4

NUCLEAR FUELS

Vessel Dome Pressure



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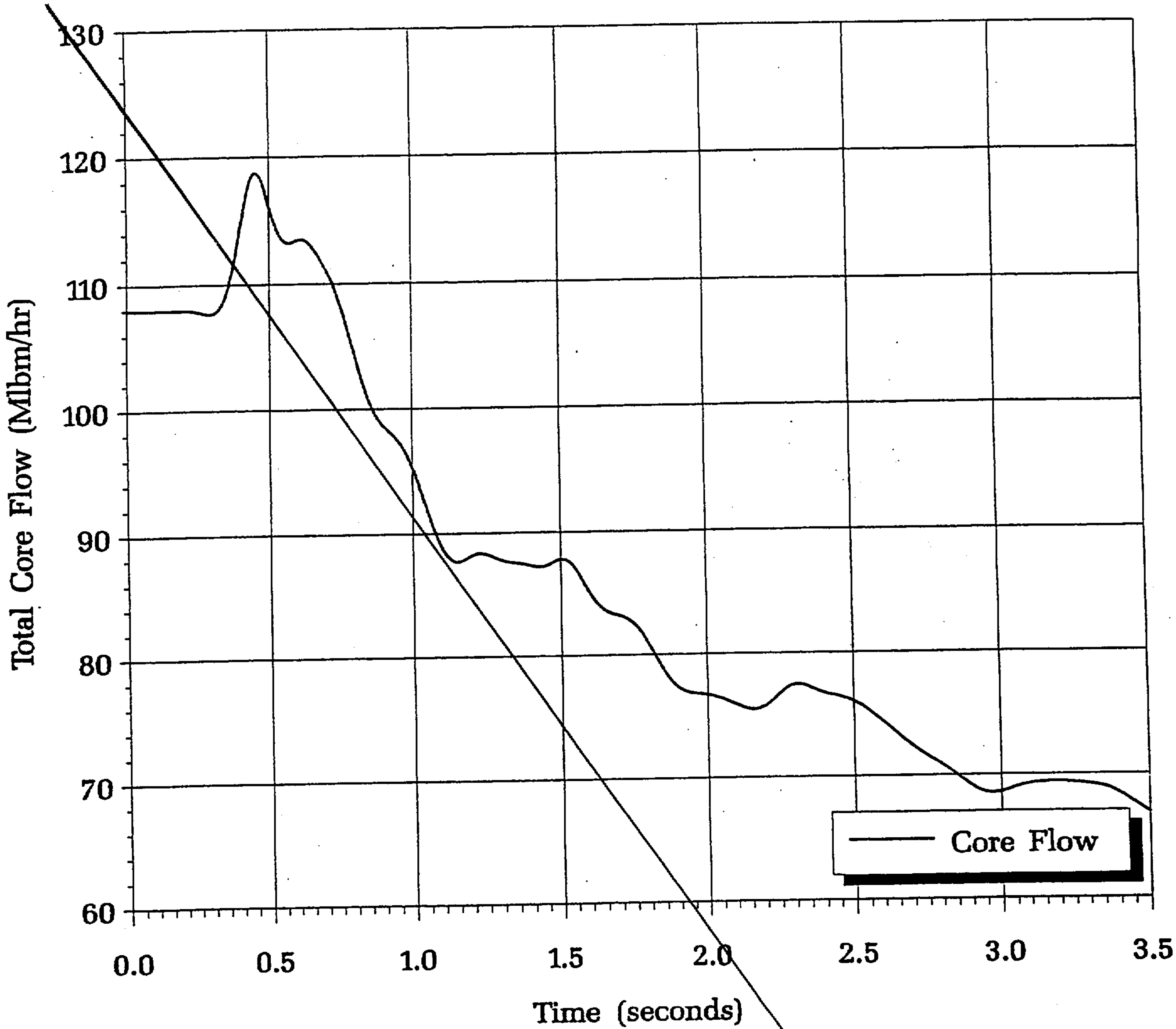
SUSQUEHANNA STEAM ELECTRIC STATION UNIT 2 CYCLE 11 FINAL SAFETY ANALYSIS REPORT
SUSQUEHANNA GENERATOR LOAD REJECT WITHOUT BYPASS AND TURBINE TRIP WITHOUT BYPASS FSAR FIGURE 15D.2.2-1-4

NUCLEAR FUELS

Total Core Flow



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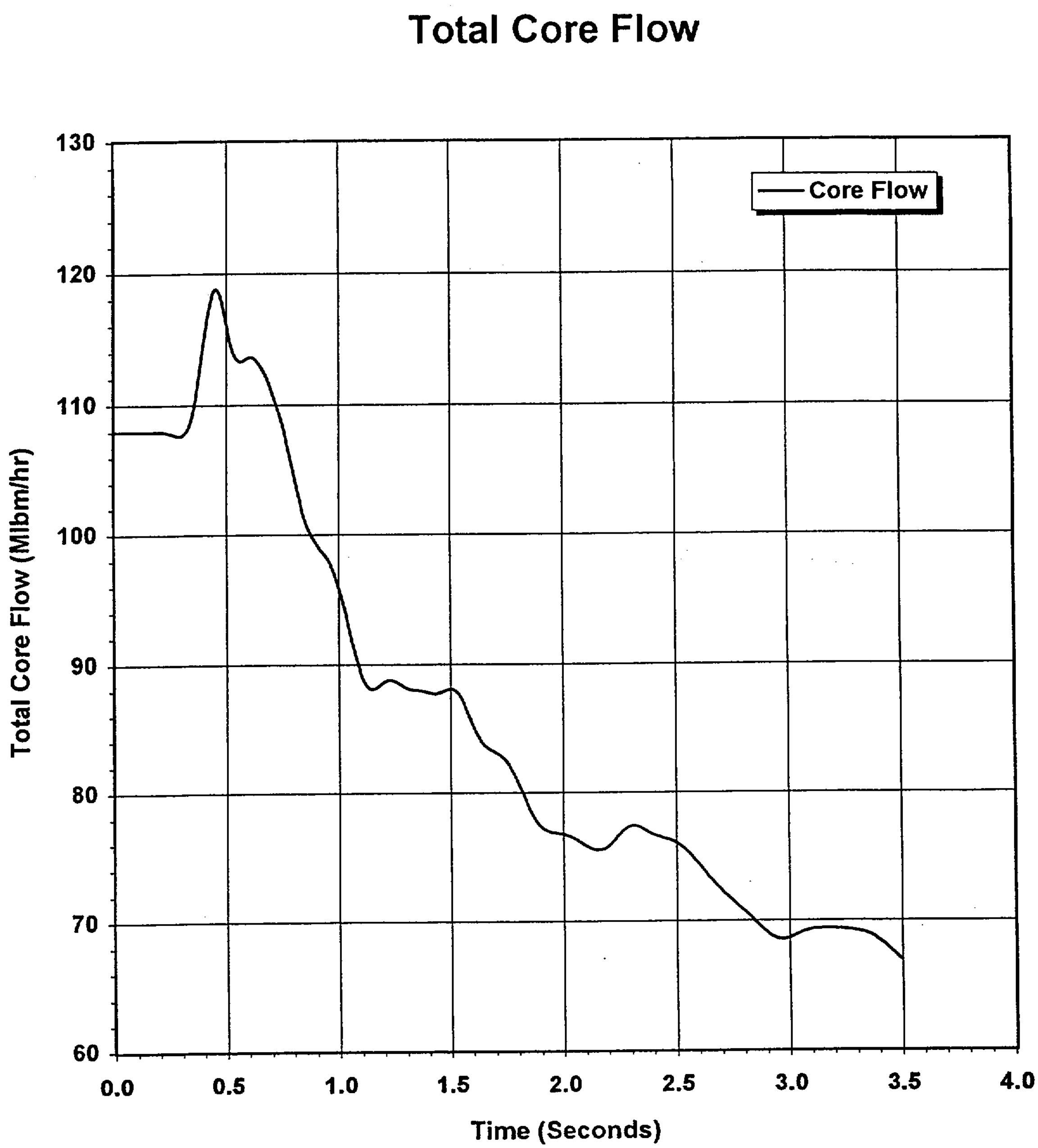
UNIT 2 CYCLE 10
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT
WITHOUT BYPASS AND
TURBINE TRIP WITHOUT BYPASS

FSAR FIGURE 15D.2.2-1-5

NUCLEAR FUELS



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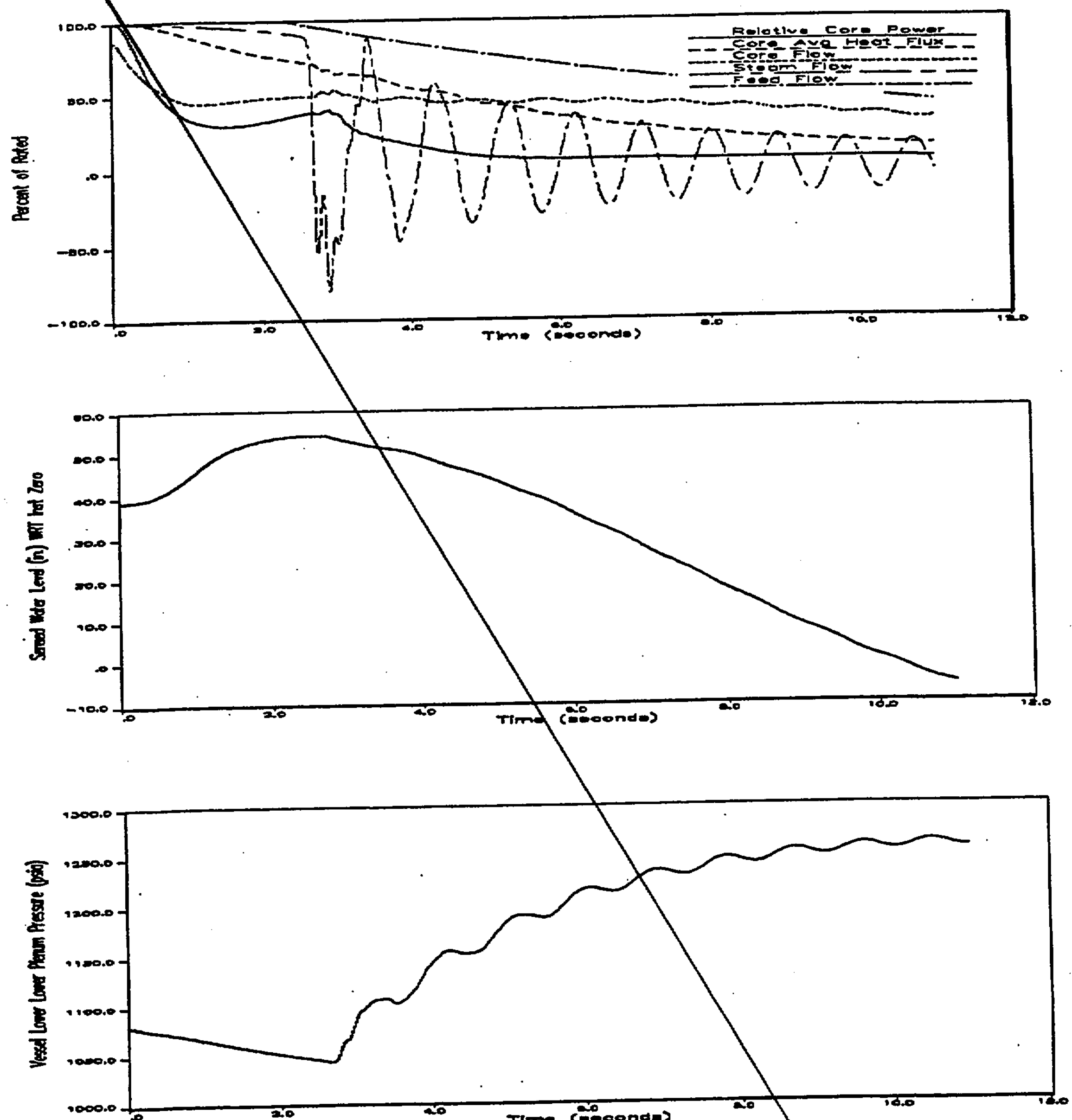
SUSQUEHANNA STEAM ELECTRIC STATION UNIT 2 CYCLE 11 FINAL SAFETY ANALYSIS REPORT
SUSQUEHANNA GENERATOR LOAD REJECT WITHOUT BYPASS AND TURBINE TRIP WITHOUT BYPASS FSAR FIGURE 15D.2.2-1-5

NUCLEAR FUELS



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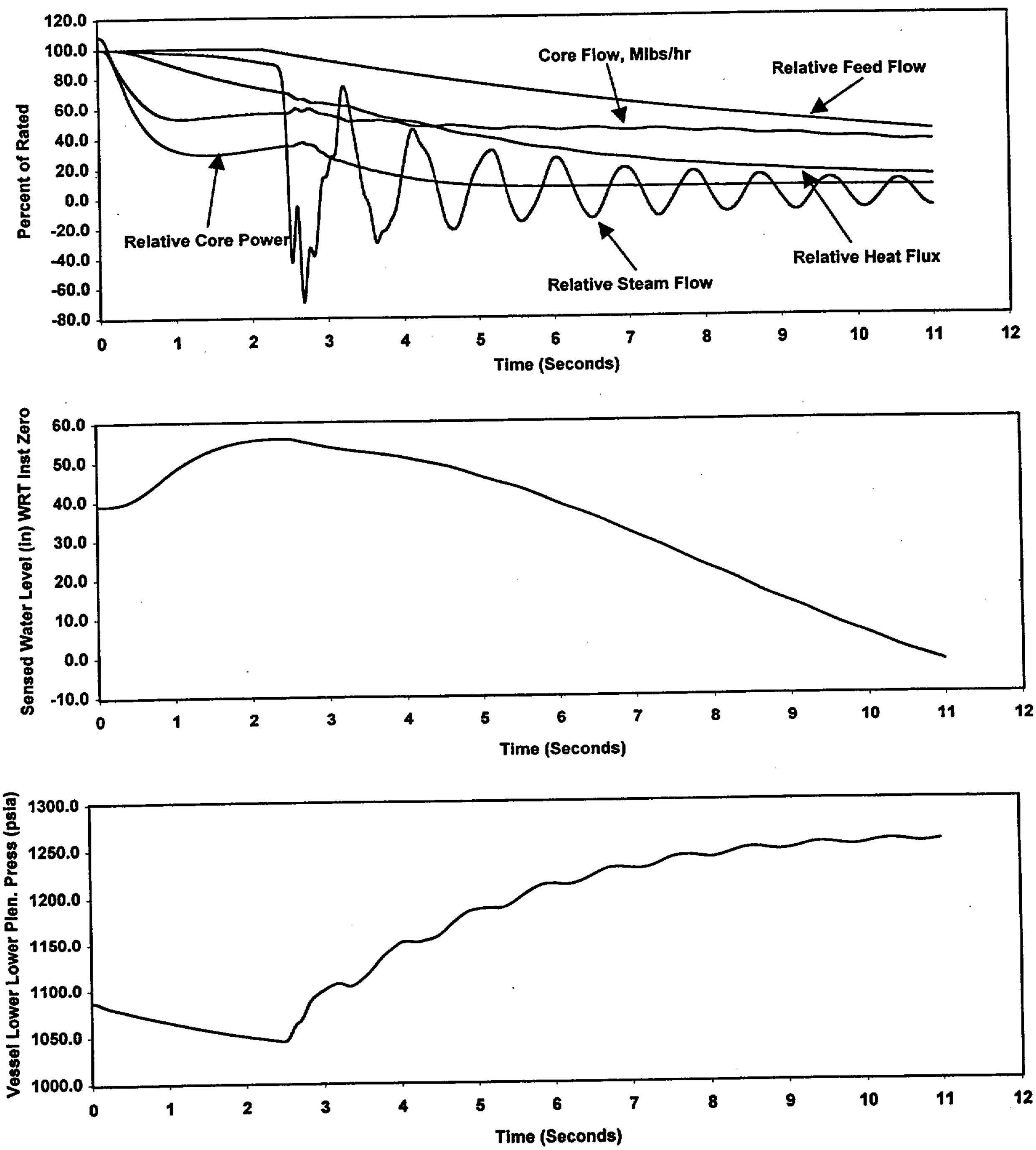
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 10
FINAL SAFETY ANALYSIS REPORT

PUMP SEIZURE ACCIDENT
TWO LOOP OPERATION
(100/87)

FSAR FIGURE 15D.3.3-1

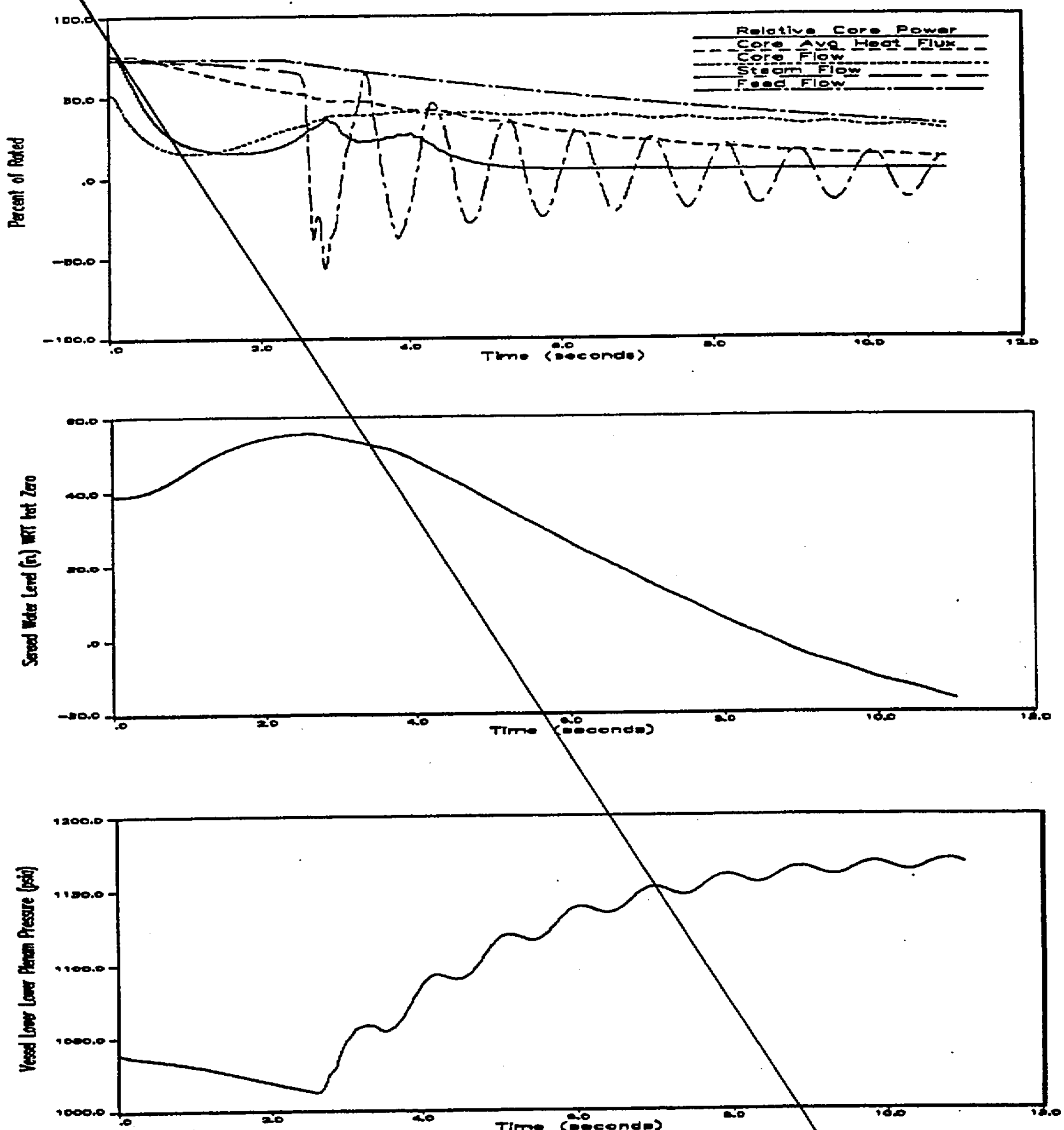
NUCLEAR FUELS



FSAR REV. 55

SUSQUEHANNA STEAM ELECTRIC STATION UNIT 2 CYCLE 11 FINAL SAFETY ANALYSIS REPORT
PUMP SEIZURE ACCIDENT TWO LOOP OPERATION (100/108) FSAR FIGURE 15D.3.3-1

NUCLEAR FUELS



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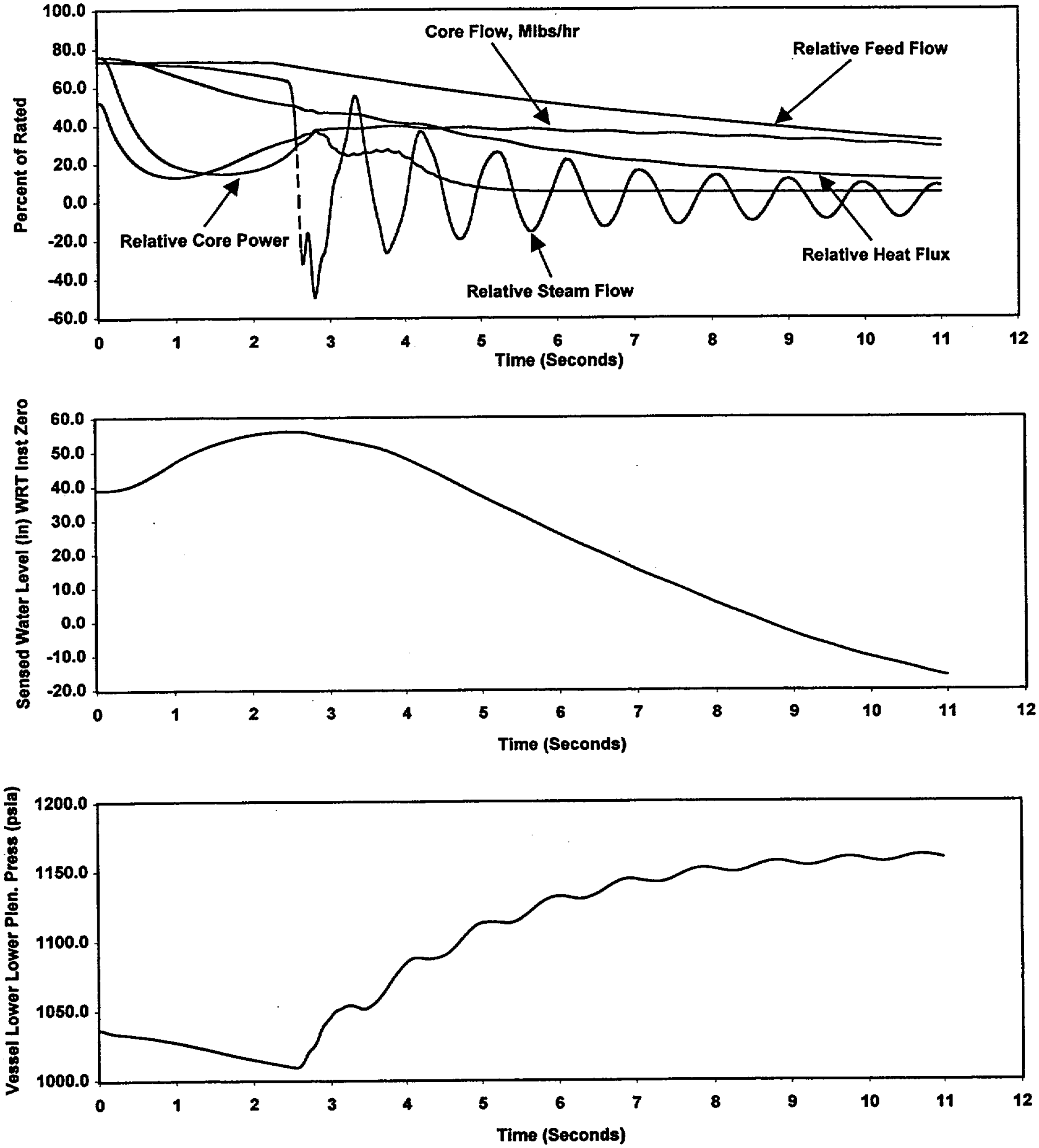
SUSQUEHANNA STEAM ELECTRIC STATION UNIT 2 CYCLE 10 FINAL SAFETY ANALYSIS REPORT
PUMP SEIZURE ACCIDENT SINGLE LOOP OPERATION (100/87) FSAR FIGURE 15D.3.3-2

NUCLEAR FUELS



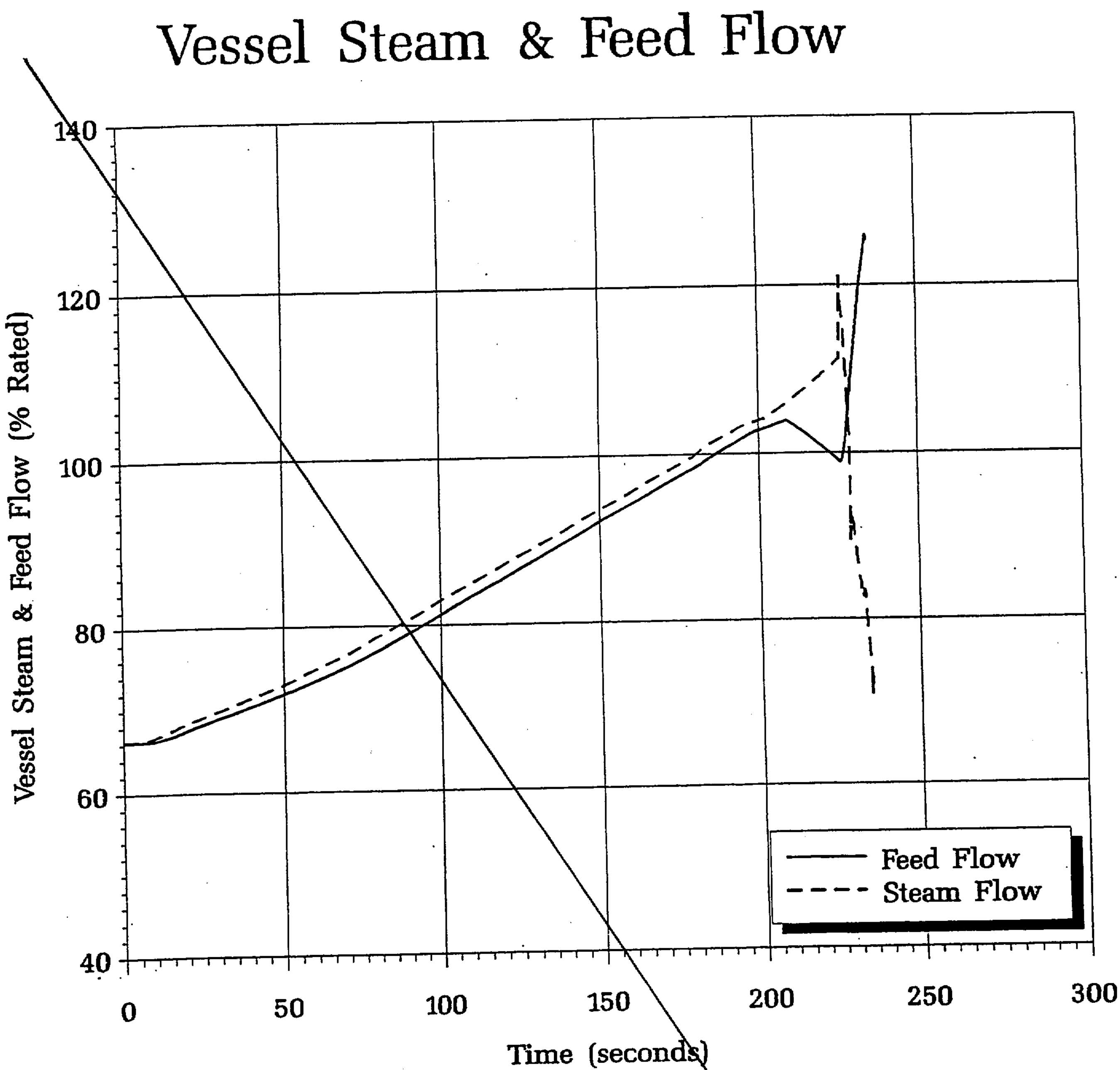
FSAR REV. 55

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 11
FINAL SAFETY ANALYSIS REPORT

PUMP SEIZURE ACCIDENT
SINGLE LOOP OPERATION
(76/52)

FSAR FIGURE 15D.3.3-2

NUCLEAR FUELS



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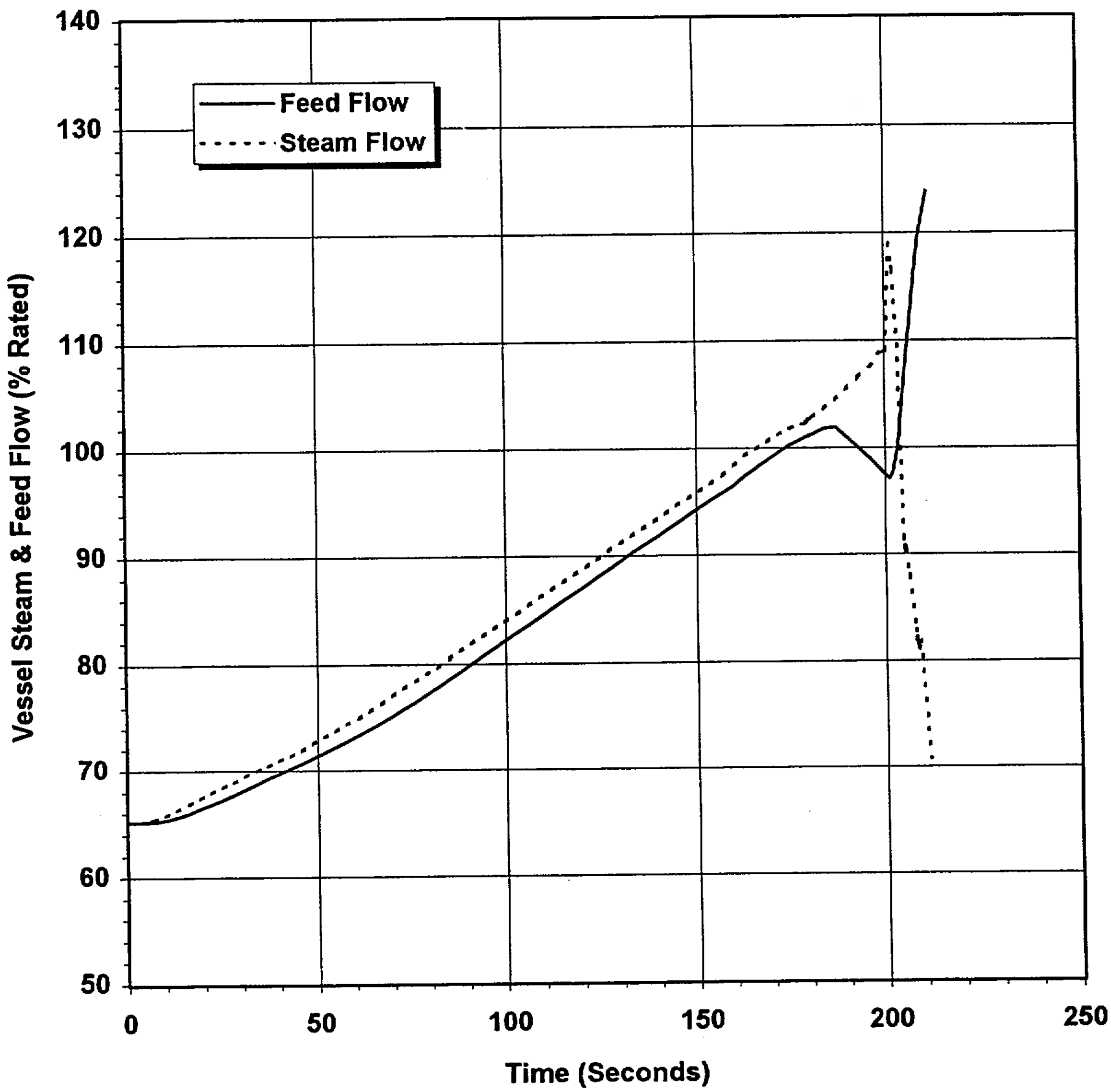
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 10
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA RECIRCULATION FLOW
CONTROLLER FAILURE
INIT. COND. 69% POWER 60 Mlbs/hr FLOW

FSAR FIGURE 15D.4.5-1-1

NUCLEAR FUELS

Vessel Steam & Feed Flow

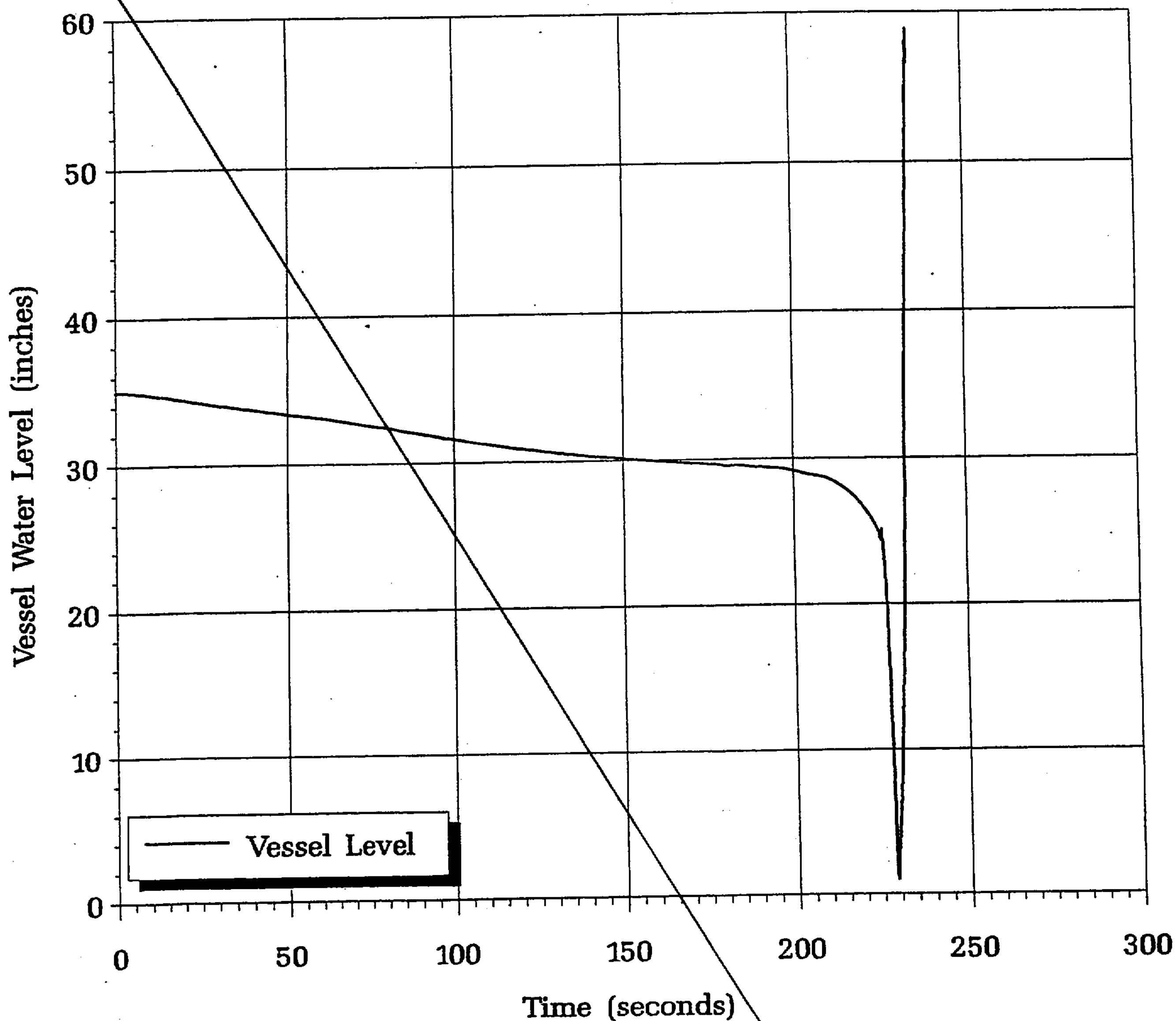


FSAR REV. 55

SUSQUEHANNA STEAM ELECTRIC STATION UNIT 2 CYCLE 11 FINAL SAFETY ANALYSIS REPORT
SUSQUEHANNA RECIRCULATION FLOW CONTROLLER FAILURE INIT. COND. 68% POWER 60 Mlbs/hr FLOW
FSAR FIGURE 15D.4.5-1-1

NUCLEAR FUELS

Vessel Water Level



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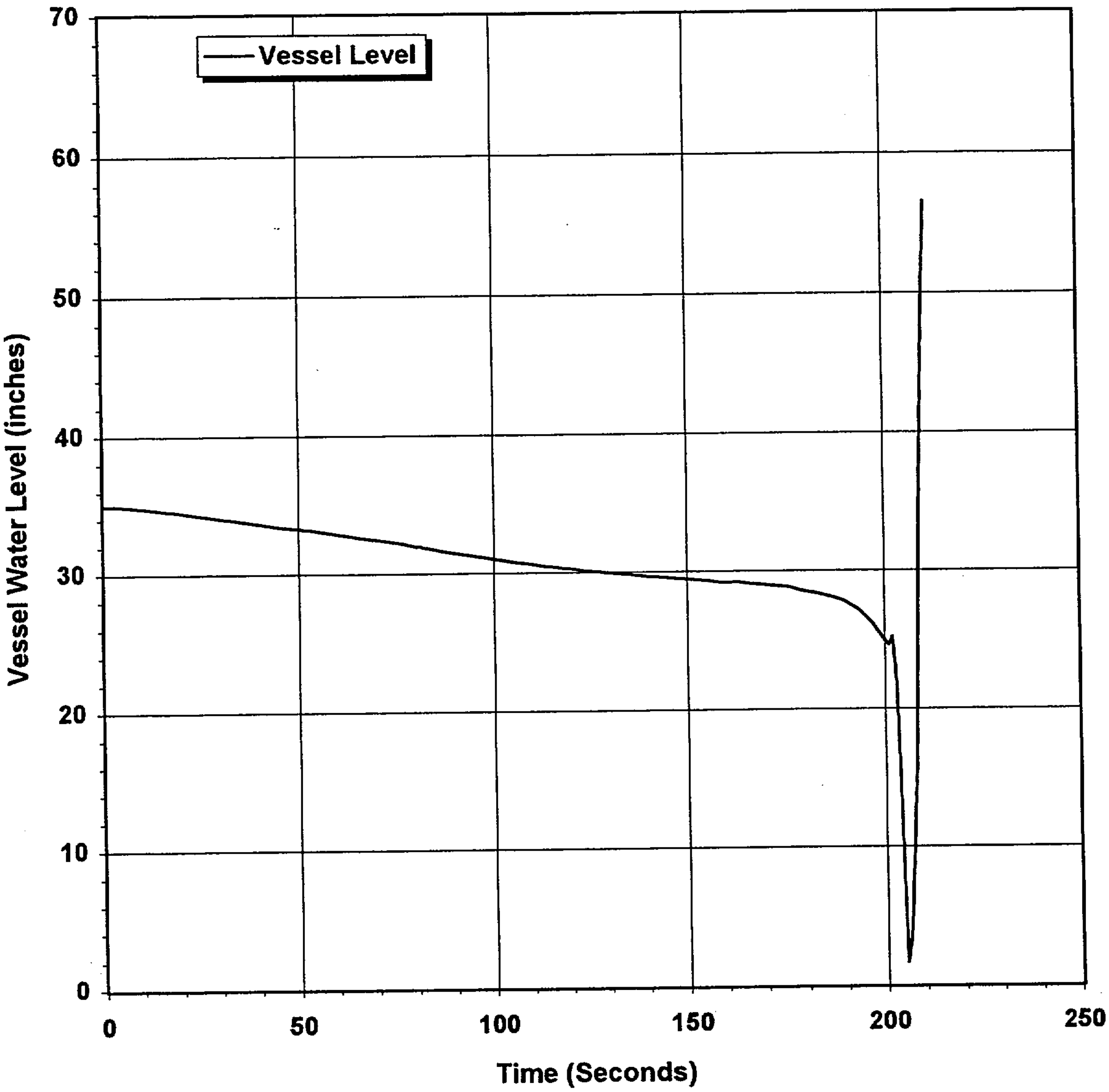
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 10
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA RECIRCULATION FLOW
CONTROLLER FAILURE
INIT. COND. 69% POWER 60 Mlbs/hr FLOW

FSAR FIGURE 15D.4.5-1-2

NUCLEAR FUELS

Vessel Water Level

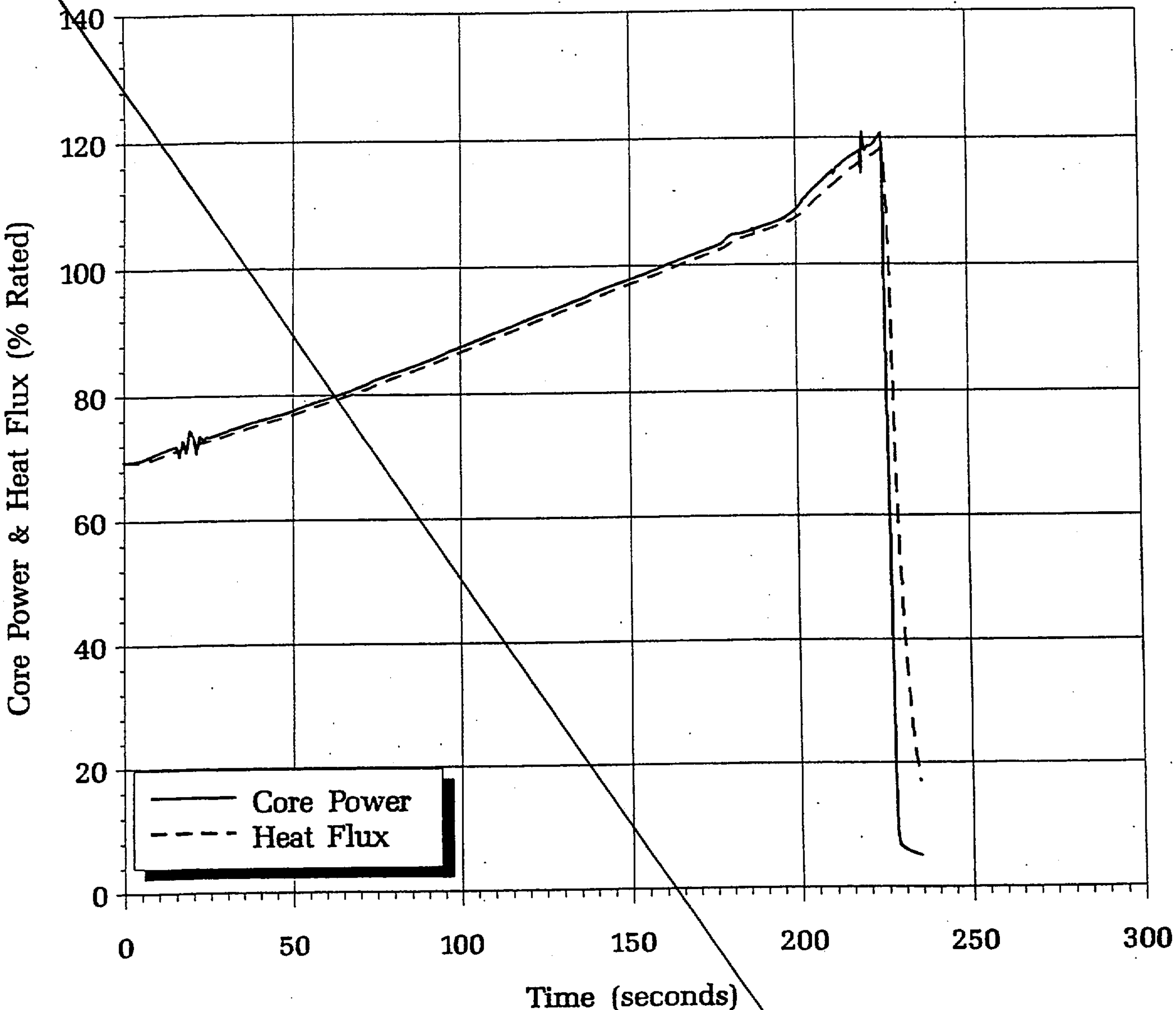


FSAR REV. 55

SUSQUEHANNA STEAM ELECTRIC STATION UNIT 2 CYCLE 11 FINAL SAFETY ANALYSIS REPORT
SUSQUEHANNA RECIRCULATION FLOW CONTROLLER FAILURE INIT. COND. 68% POWER 60 Mlbs/hr FLOW FSAR FIGURE 15D.4.5-1-2

NUCLEAR FUELS

Core Power & Heat Flux



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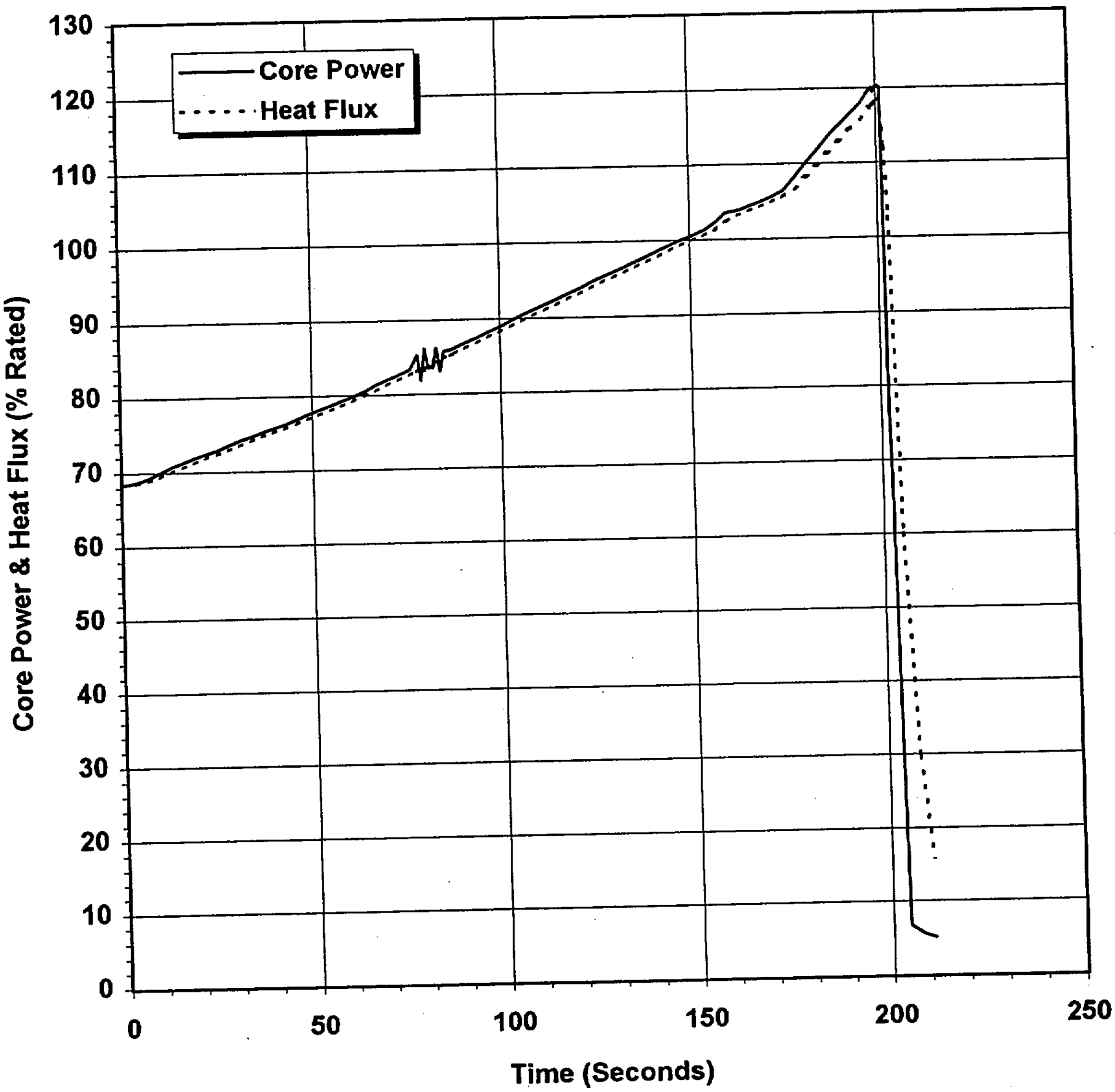
SUSQUEHANNA STEAM ELECTRIC STATION UNIT 2 CYCLE 10 FINAL SAFETY ANALYSIS REPORT
SUSQUEHANNA RECIRCULATION FLOW CONTROLLER FAILURE INIT. COND. 69% POWER 60 Mlbs/hr FLOW FSAR FIGURE 15D.4.5-1-3

NUCLEAR FUELS

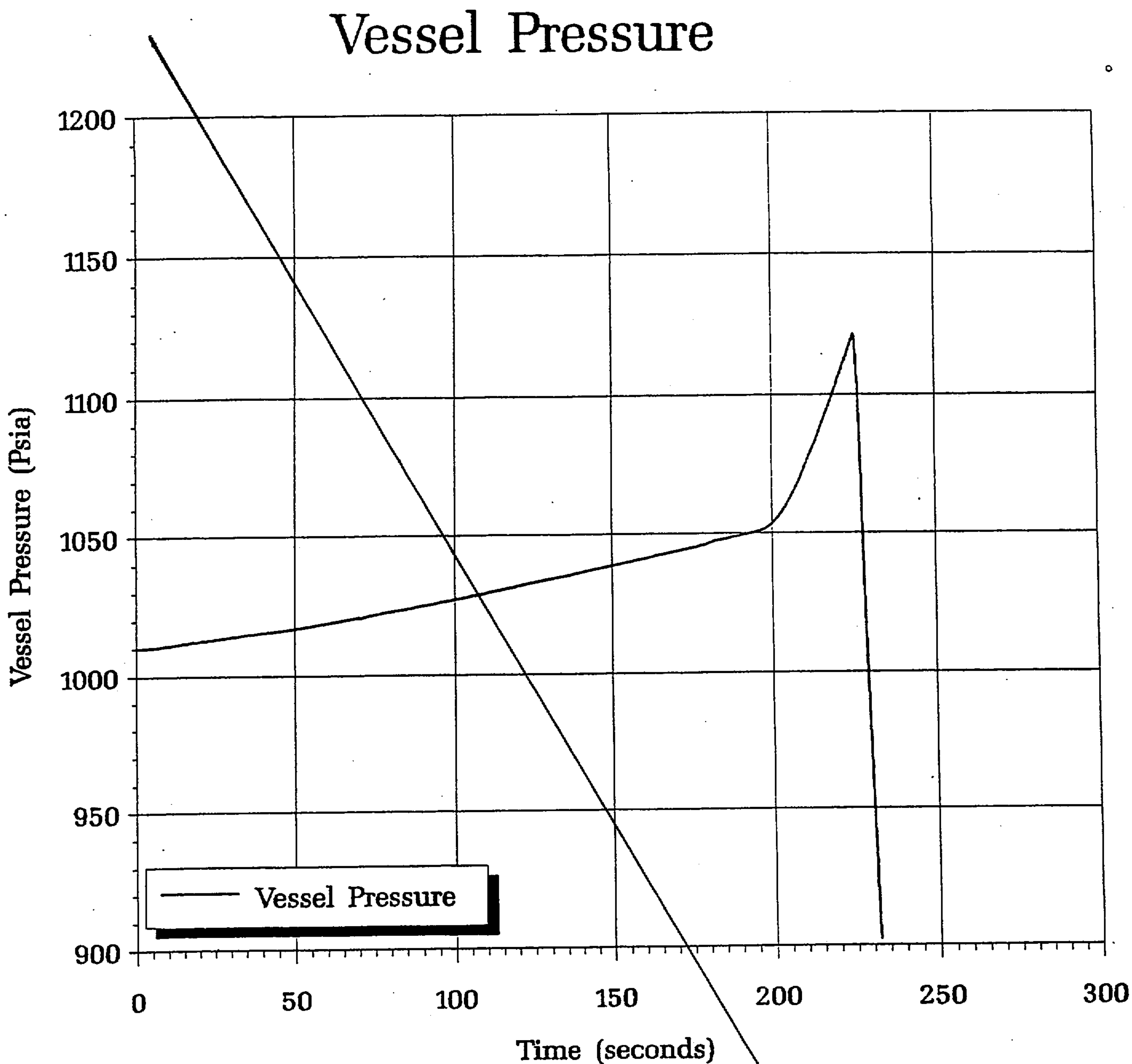
Core Power & Heat Flux



FSAR REV. 55

SUSQUEHANNA STEAM ELECTRIC STATION UNIT 2 CYCLE 11 FINAL SAFETY ANALYSIS REPORT
SUSQUEHANNA RECIRCULATION FLOW CONTROLLER FAILURE INIT. COND. 68% POWER 60 Mlbs/hr FLOW FSAR FIGURE 15D.4.5-1-3

NUCLEAR FUELS



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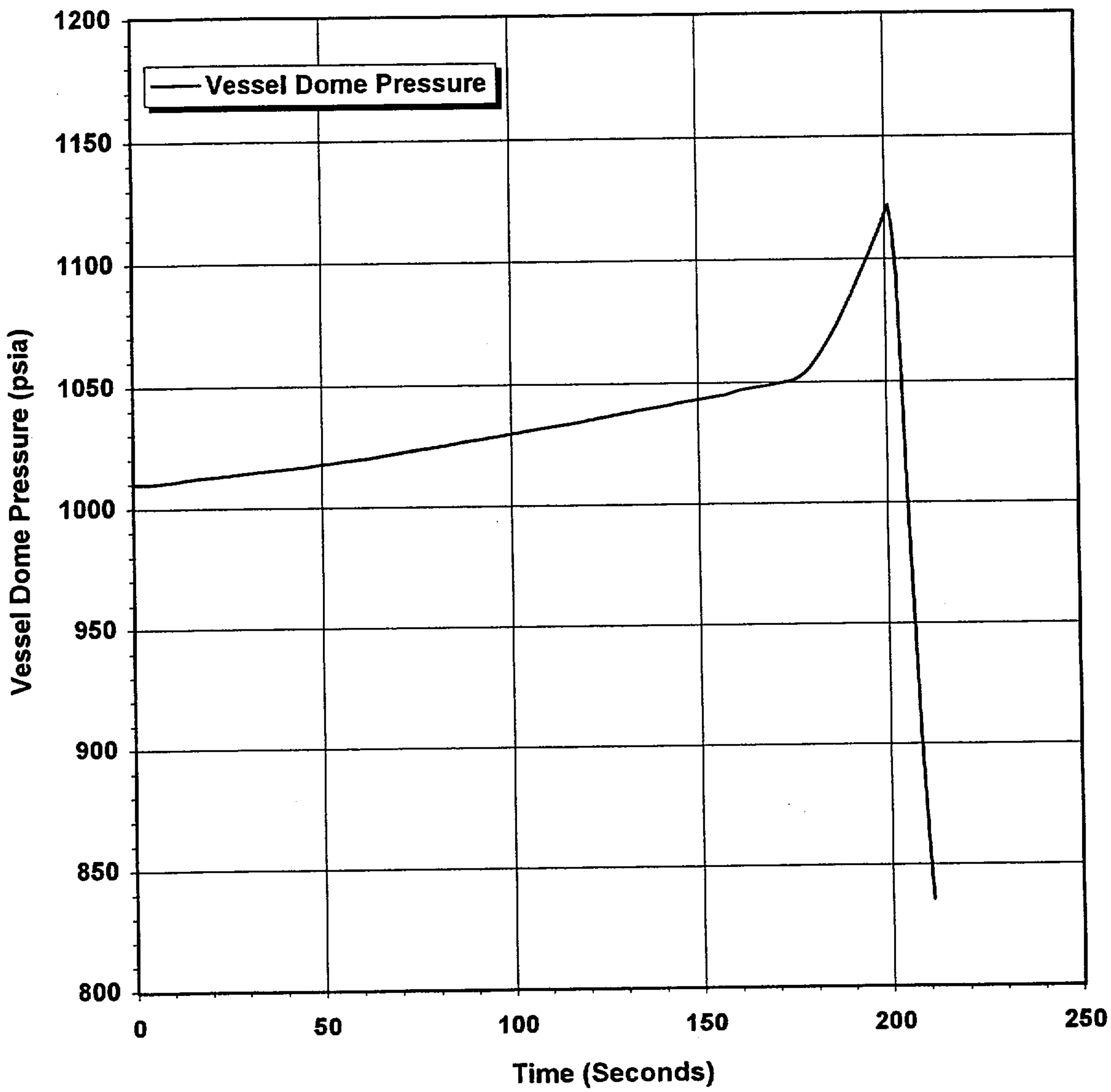
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 10
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA RECIRCULATION FLOW
CONTROLLER FAILURE
INIT. COND. 69% POWER 60 Mlbs/hr FLOW

FSAR FIGURE 15D.4.5-1-4

NUCLEAR FUELS

Vessel Dome Pressure



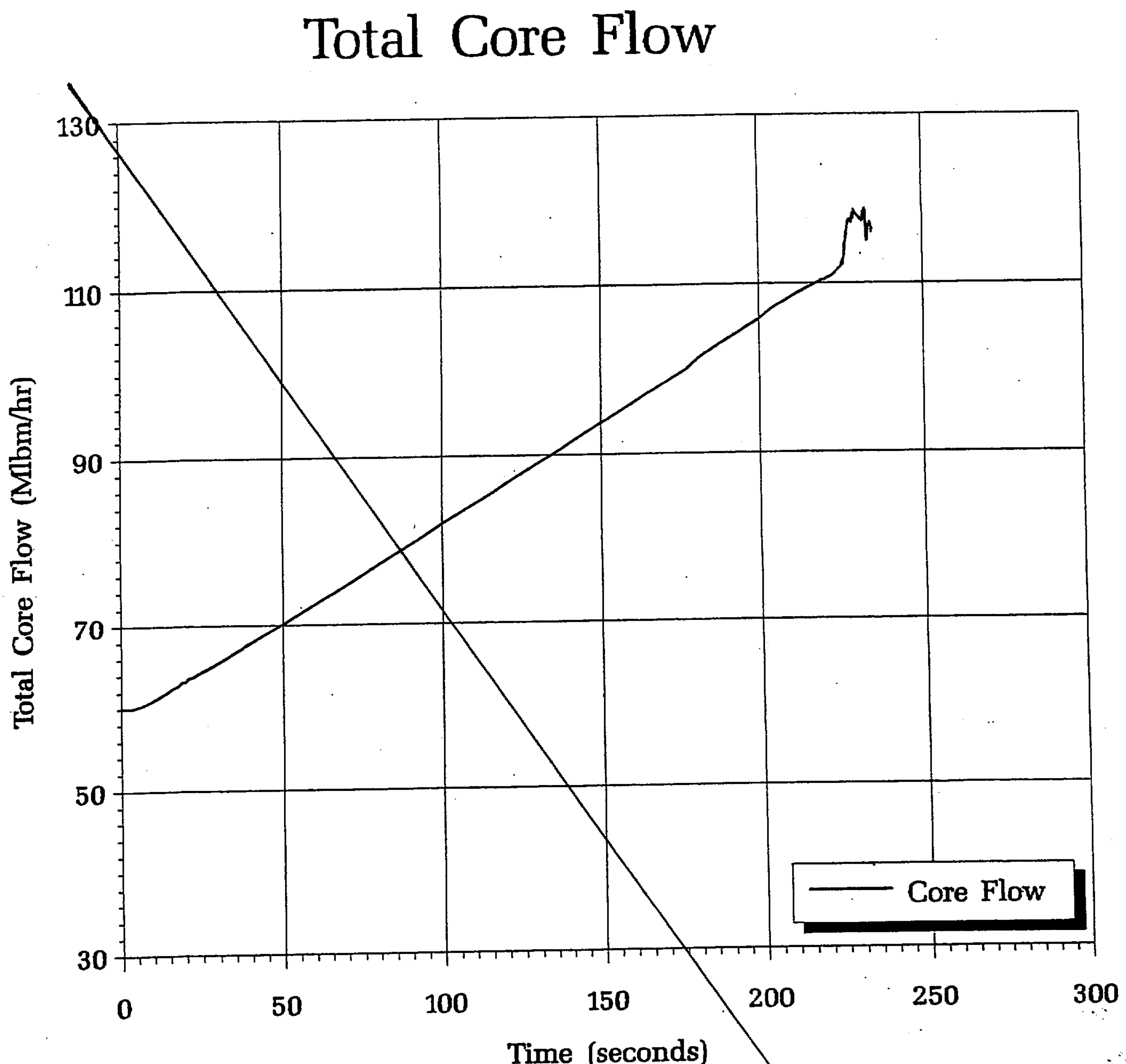
FSAR REV. 55

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 11
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA RECIRCULATION FLOW
CONTROLLER FAILURE
INIT. COND. 68% POWER 60 Mlbs/hr FLOW

FSAR FIGURE 15D.4.5-1-4

NUCLEAR FUELS



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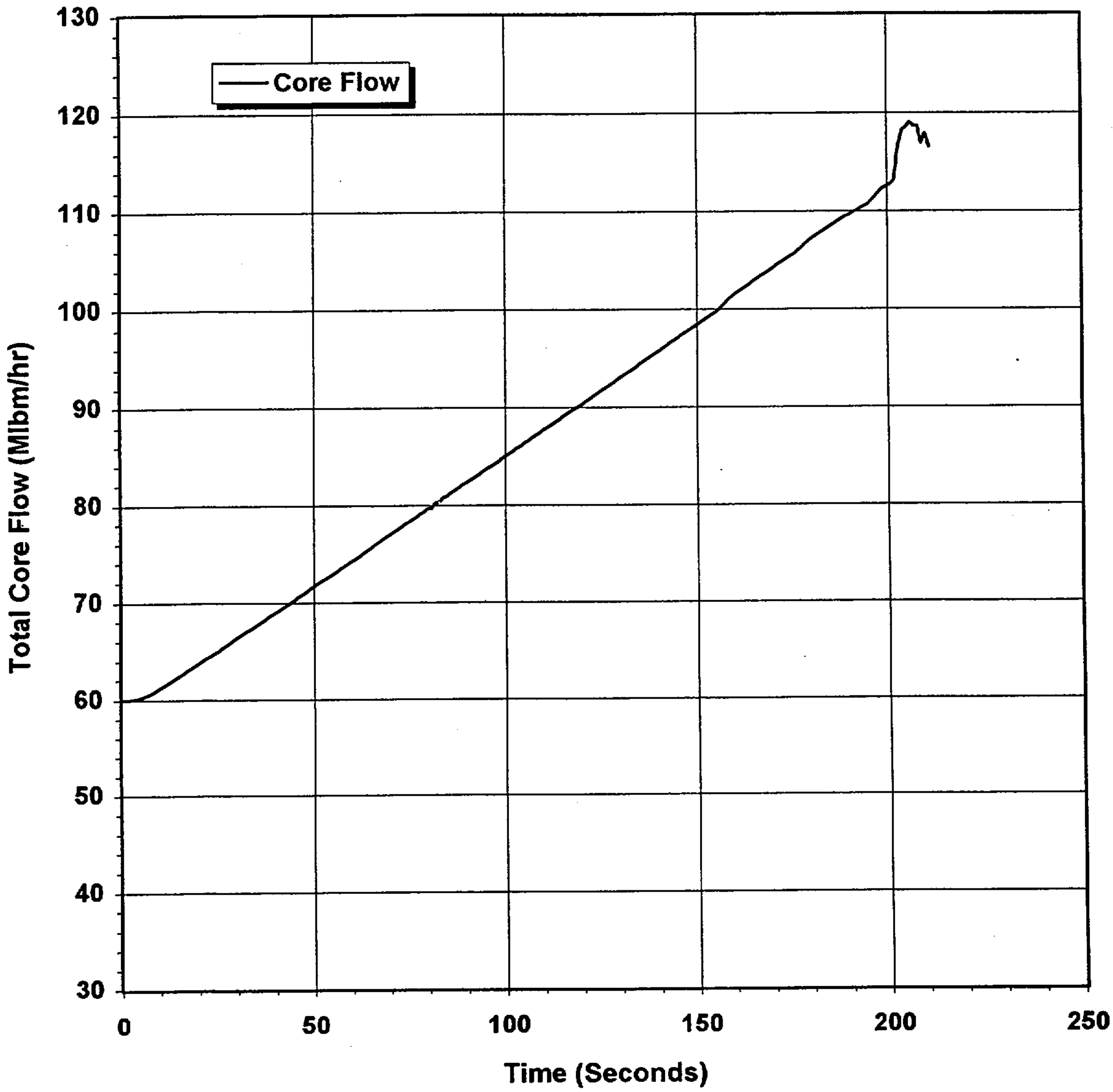
SUSQUEHANNA STEAM ELECTRIC STATION UNIT 2 CYCLE 10 FINAL SAFETY ANALYSIS REPORT
SUSQUEHANNA RECIRCULATION FLOW CONTROLLER FAILURE INIT. COND. 69% POWER 60 Mlbs/hr FLOW FSAR FIGURE 15D.4.5-1-5

NUCLEAR FUELS

Total Core Flow



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SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 CYCLE 11
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA RECIRCULATION FLOW
CONTROLLER FAILURE
INIT. COND. 68% POWER 60 Mlbs/hr FLOW

FSAR FIGURE 15D.4.5-1-5

NUCLEAR FUELS

PLA-2852
Unit 2 In-Core Nuetron Dosimetry

G3709-1687 TAH A:

PP&L Pennsylvania Power & Light Company
Two North Ninth Street • Allentown, PA 18101 • 215 / 770-5151

Harold W. Keiser
Vice President-Nuclear Operations
215/770-7502

MAY 0 8 1987

Director of Nuclear Reactor Regulation
Attention: Dr. W. R. Butler, Project Director
Project Directorate I-2
Division of Reactor Projects
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2 IN-CORE NEUTRON DOSIMETRY
PLA-2852 FILES R41-2, A17-16

Dear Dr. Butler:

The purpose of this letter is to document Pennsylvania Power & Light Company's position with regard to the missing neutron dosimeter in the Unit 2 reactor pressure vessel. What follows is a description of the event.

As part of the ISI program, the separate GE neutron dosimeter at the 30° azimuth was to be removed and analyzed. The camera crew could not locate the dosimeter however they did locate the dosimeter holder which was empty and appeared undamaged. An NCR was written documenting the nonconforming condition and the potential for a loose part in the vessel. Additional, unsuccessful underwater camera inspections were conducted in an attempt to locate the dosimeter.

The construction documentation was reviewed and it was determined that the dosimeter was to have been installed in April, 1983 but there was no evidence to substantiate whether or not the dosimeter was ever installed. PP&L's Procurement Department was requested to obtain a dosimeter from another utility - which they did - and in parallel, plans were made for its installation. Prior to any installation attempts, PP&L received information which negated the need to install the dosimeter. Throughout this period of time, NRC's resident inspector was kept informed with regard to the missing dosimetry.

It is PP&L's position that installation of a neutron dosimeter is neither necessary or beneficial, nor does the lack thereof impact plant safety. Data from the dosimeter would have been used to verify the fluence estimate used to generate the P-T curves for Unit 2. Since both Susquehanna units have geometric similarity, the Unit 1 data is applicable to Unit 2. The Unit 1 data shows predicted values for neutron fluence to be 1.75 times greater than actual conditions. Information obtained from GE indicates that other BWRs measured fluences are less than or equal to estimated values.

Also, installation of the neutron dosimeter at this time would have little benefit to Susquehanna Unit 2. The Unit 2 P-T curves are not beltline (neutron embrittlement) limited, therefore there would be no potential for revising the curves on the basis of dosimeter data. When Regulatory Guide 1.99, Revision 2 is issued, the beltline will become limiting however the impact on Unit 2 is minor - the 1100 psig minimum hydrotest temperature increases from 170°F to 176°F.

PP&L also maintains that Susquehanna Unit 2 is currently in compliance with 10CFR50 Appendix G, 10CFR50 Appendix 4, and ASTM-E-185-73 since there are neutron dosimeters within each surveillance specimen capsule as required by the regulations. Also, the dosimeter in the lead capsule (to be withdrawn in 6 EFPY) will not saturate and will be usable to determine neutron fluence.

Proposed FSAR changes to reflect the missing dosimetry are attached.

If you have any questions, please contact us.

Very truly yours,

H Keiser

H. W. Keiser
Vice President - Nuclear Operations

Attachments

cc: NRC Document Control Desk (original)
NRC Region I
Mr. L. R. Plisco, NRC Resident Inspector
Mr. M. C. Thadani, NRC Project Manager

bcc:		
	C. T. Coddington	A2-4
	E. A. Heckman	A2-4
	D. J. Morgan	A6-2
	D. J. Walters	A2-4
	SRMS Corresp. File	A6-2

DJW:tah
djwmed194a

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5.3.1.6 Material Surveillance

5.3.1.6.1 Compliance with "Reactor Vessel Material Surveillance Program Requirements"

The materials surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from their exposure to neutron irradiation and thermal environment.

Materials for the program are selected to represent materials used in the reactor beltline region. The specimens are manufactured from a plate actually used in the beltline region and a weld typical of those in the beltline region and thus represent base metal, weld metal, and the transition zone between base metal and weld. The plate and weld are heat treated in a manner which simulates the actual heat treatment performed on the core region shell plates of the completed vessel.

REPLACE WITH ATTACHED

~~The surveillance program includes three capsule holders per reactor vessel. Each holder is loaded with capsules which contain the following surveillance specimens:~~

~~First Holder~~	-	~~36 Charpy impact specimens which consist of 12 base metal, 12 weld metal, and 12 weld heat affected zone material; 10 tensile specimens which consist of 3 base metal, 4 weld metal, and 3 weld heat affected zone material.~~
~~Second Holder~~	-	~~24 Charpy impact specimens which consist of 8 base metal, 8 weld metal, and 8 weld heat affected zone material; 6 tensile specimens which consist of 2 base metal, 2 weld metal, and 2 weld heat affected zone material.~~
~~Third Holder~~	-	~~24 Charpy impact specimens which consist of 8 base metal, 8 weld metal, and 8 weld heat affected zone material; 8 tensile specimens which consist of 3 base metal, 3 weld metal, and 2 weld heat affected zone material.~~

A set of out-of-reactor baseline Charp V-notch specimens is provided with the surveillance test specimens.

Charpy impact specimens for the reactor vessel surveillance programs are of the longitudinal orientation consistent with the ASME requirements prior to the issuance of the Summer 1972 Addenda and ASTM-E-185-73. Based on GE experience, the amount of

Present form.

shift measured by these irradiated longitudinal test specimens will be essentially the same as the shift in an equivalent transverse specimen.

The program includes three capsules in the reactor. Since the predicted adjusted reference temperature of the reactor vessel beltline steel is less than 100°F at end of life, the use of three capsules meets the requirements of 10 CFR 50, Appendix H, and ASTM-E-185-73. The withdrawal schedule is provided in the Technical Specifications for each unit.

For the extent of compliance to 10 CFR 50, Appendix H, see Tables 5.3-1b and 5.3-2b.

5.3.1.6.2 Neutron Flux and Fluence Calculations

A description of the methods of analysis is contained in Subsections 4.1.4.5 and 4.3.2.8.

5.3.1.6.3 Positioning of Surveillance Capsules and Method of Attachment

Surveillance specimen capsules are located at three azimuths at a common elevation in the core beltline region. The sealed capsules are not attached to the vessel but are in welded capsule holders. The capsule holders are mechanically retained by capsule holder brackets welded to the vessel cladding as shown in Figure 5.3-3. The capsule holder brackets allow the capsule holder to be removed at any desired time in the life of the plant for specimen testing. These brackets are designed, fabricated and analyzed to the requirements of Section III of the ASME Code. A positive spring-loaded locking device is provided to retain the capsules in position throughout any anticipated event during the lifetime of the vessel.

5.3.1.6.4 Time and Number of Dosimetry Measurements

~~GE provides a separate neutron dosimeter so that fluence measurements may be made at the vessel ID during the first fuel cycle to verify the predicted fluence at an early date in plant operation. This measurement is made over this short period to avoid saturation of the dosimeters now available. Once the fluence-to-thermal power output is verified, no further dosimetry is considered necessary because of the linear relationship between fluence and power output.~~

SSES FSAR Section 5.3.1.6 Revisions Proposed

5.3.1.6.1

The surveillance program includes three capsule holders per reactor vessel. Each holder is loaded with capsules which contain the following surveillance specimens and dosimeter wires:

First holder:

36 Charpy impact specimens including 12 base metal, 12 weld metal, and 12 heat affected zone metal specimens; 10 tensile specimens including 3 base metal, 4 weld metal, and 3 weld heat affected zone metal specimens; 4 metal wire dosimeters including 2 made of iron and 2 made of copper.

Second holder:

24 Charpy impact specimens including 8 base metal, 8 weld metal, and 8 weld heat affected zone metal specimens; 6 tensile specimens including 2 base metal, 2 weld metal, and 2 weld heat affected zone metal specimens; 4 metal wire dosimeters including 2 made from iron and 2 made from copper.

Third Holder:

24 Charpy impact specimens including 8 base metal, 8 weld metal and 8 weld heat affected zone metal specimens; 8 tensile specimens including 3 base metal, 3 weld metal, and 2 weld heat affected zone metal specimens; 4 metal wire dosimeters including 2 made of iron and 2 made of copper.

Proposed Revision to Section 5.3.1.6.4:

5.3.1.6.4 Time and Number of Dosimeter Measurements

GE has provided neutron dosimetry wires in each of the specimen holders. The first holder removed will have its wires analyzed, the neutron fluence calculated and the result compared to the predicted values. No further dosimetry is considered necessary because of the linear relationship between fluence and power output. The capsule withdrawal schedule is listed in the Technical Specifications, Table 4.4.6.1.3-1.

Calculation EC-031-1010

32

NUCLEAR ENGINEERING
CALCULATION / STUDY COVER SHEET and
NUCLEAR RECORDS TRANSMITTAL SHEET

➢2. TYPE: CALC ➢3. NUMBER: EC-031-1010 ➢4. REVISION: 0

5. TRANSMITTAL#: K0007021 *➢6. UNIT: 3 *➢7. QUALITY CLASS: N

➢9. DESCRIPTION: Determination of Error Band on Reactor Thermal Power *➢8. DISCIPLINE: D

Calculation - Before and After Feedwater Flow Element Upgrade

SUPERSEDED BY: EC-

10. Alternate Number: 11. Cycle: N/A

12: Computer Code or Model used: None Fiche ☐ Dis ☐ Am't

13. Application: Uncertainty determineation for core thermal power calculation on PICSY

*➢14 *Affected Systems:* 031 , 045 , 062 , 064 , ,

* If N/A then line 15 is mandatory.

**➢15. NON-SYSTEM DESIGNATOR: THYD

**If N/A then line 14 is mandatory

16. Affected Documents: EC-031-1008; NE-2000-001 (Licensing Topical Report supporting LEFM™ Upgrade) ☐ SAR Change Req'd

17. References: See Reference Section (Page 32)

18. Equipment / Component #: Replacement LEFM™ flow elements for Feedwater flow measurement

19. DBD Number: DBD030; DBD039

➢20. PREPARED BY	J. G. Refling *Print Name*	Jack G. Refling 6/27/00 *Signature*
➢21. REVIEWED BY	M. A. Adelizzi *Print Name*	Michael A Adelizzi 6/27/00 *Signature*
➢21A. VERIFIED BY	*Print Name*	*Signature*
➢22. APPROVED BY	F. G. Butler *Print Name*	F G Butler 6/28/00 *Signature*
➢23. ACCEPTED BY PP&L / DATE	*Print Name*	*Signature / DATE*

TO BE COMPLETED BY NUCLEAR RECORDS

NR-DCS SIGNATURE/DATE Ruth C Chanity

ADD A NEW COVER PAGE FOR EACH REVISION
FORM NEPM-QA-0221-1, Revision 3, Page 1 of 2, ELECTRONIC FORM

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Introduction

Pennsylvania Power and Light Company (ppl) is in the process of installing new flow measurement devices for determining feedwater flow at Susquehanna Steam Electric Station (SSES). The flow measurement devices to be installed use the Leading Edge Flow Meter™ (LEFM✓™) technology developed by Caldon Corporation[1]. The basis for the installation of the LEFM✓™ flow meters is their increased accuracy in the measurement of core thermal power, which then allows for increasing the rated thermal power of the SSES units. The purpose of this calculation is to estimate the accuracy of the present core thermal power measurement and compare that accuracy with the computed accuracy of the LEFM✓™ system

The Nuclear Regulatory Commission (NRC), in Regulatory Guide 1.49[2], established that, for design and accident analysis purposes, the licensed core thermal power level (or rated thermal power (RTP)) must be increased by at least 2% to allow for instrumentation inaccuracies in the calculation of RTP. This 2% increase in actual power level assumed in design and accident analyses was a 95% confidence limit, that is, the 102% power level was chosen to be a core thermal power level that would provide an upper bound for actual core thermal power at least 95% of the time the plant is operating. Currently, the RTP at SSES is 3441 MWt, and the design bases power level is 3510 MWt, or 1.02 times the RTP.

The LEFM✓™ system is reported to reduce the measurement inaccuracy of the RTP calculation from the Regulatory Guide mandated 2% to less than 0.6%. Based on this reported value and the NRC acceptance of a submittal by Texas Utilities[3] to increase RTP based on the reduction in RTP determination, Ppl is installing the LEFM✓™ flow measurement system to measure feedwater flow. The feedwater flow measurement contributes the single most important portion of the inaccuracy to the RTP calculation, as will be demonstrated below. Therefore, increasing the accuracy of the feedwater flow measurement accuracy allows for the increase in RTP, without sacrificing any design margin. PPL is petitioning the NRC to increase RTP by 1.4% on the basis of the increased RTP measurement accuracy.

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1 'TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM™ System' CALDON, Incorporated Engineering Report 80P, Revision 0, March 1997.

2 United States Atomic Energy Commission, Regulatory Guide 1.49, 'Power Levels of Nuclear Power Plants,' Rev. 1, December 1973.

3 NRC SER on Caldon Topical and TU Submittal

Conclusions

The methodology and calculations presented below show that the overall uncertainty in the calculation of total core thermal power is currently ± 0.843% on a 95% confidence interval. This uncertainty is made up of essentially two terms, the uncertainty in feedwater flow determination and all other uncertainties. For the current situation, the ratio of uncertainty in feedwater flow uncertainty to overall uncertainty is

$$\frac{(0.792)}{(0.835)} = 0.95,$$

or 95% of the uncertainty in the calculation for total core thermal power results from the uncertainty in feedwater flow measurement.

The implementation of the LEFM✓™ feedwater flow measurement system reduces the uncertainty in feedwater flow measurement to 0.437% and the uncertainty in the total core thermal power calculation to 0.546%. The ratio of uncertainty in the calculation of feedwater flow to the uncertainty in the total core thermal power calculation is then

$$\frac{(0.438)}{(0.551)} = 0.795,$$

or 79.5% of the total core thermal power calculation uncertainty now comes directly from the uncertainty in the feedwater flow calculation.

Based on the results in the following section, it can be concluded that

1. The current total core thermal power calculation uncertainty is ± 0.843%, which is well within the NRC assumption in total core thermal power calculation (±2%).
2. The uncertainty in total core thermal power calculated with the implementation of the modification to add the LEFM✓™ flow measurement system for the determination of feedwater flow is less than ± 0.6%. Therefore, an increase in rated core thermal power of 1.4% (that is, 2% - 0.6%) is justified, since the probability of exceeding the analysis limit on core thermal power is not changed.

Methodology

A. Current Instrumentation

The reactor heat balance currently used at SSES is[4]:

$$Q_{TP} = Q_{RAD} + Q_{FW} + Q_{CRD} + Q_{CU} + Q_{PP} - Q_P - Q_{SAMPLE}$$

where Q_{TP} = total thermal power generated by the reactor core, MWt

Q_{RAD} = Ambient heat losses to drywell and piping, MWt

Q_{FW} = Power transferred to feedwater, MWt

$$= \frac{(\dot{M}_{FWA}(h_s - h_{FWA}) + \dot{M}_{FWB}(h_s - h_{FWB}) + \dot{M}_{FWC}(h_s - h_{FWC}))}{Const}$$

h_s = steam enthalpy, Btu/ lb_m

h_{FW} = feedwater enthalpy, Btu/ lb_m

$\dot{M}_{FW}$ = mass flow rate, lb_m/hr

A,B,C = subscripts indicating one of the three feedwater loops

$Const$ = Conversion from Btu/hr to MW = 3.41214163×10^6 Btu / MW-hr

Q_{CRD} = Power transferred to control rod drive system fluid, MW

$$= \frac{\dot{M}_{CRD}(h_s - h_{CRD})}{Const}$$

$\dot{M}_{CRD}$ = Control rod drive system flow rate, lb_m/hr

h_{CRD} = Enthalpy of the CRD system fluid, Btu/lb_m

Q_{CU} = Net energy loss in the reactor water clean-up (RWCU) system, MW

$$= \left[\frac{\dot{M}_{NLF51}(h_{NLT52} - h_{NLT51}) + \dot{M}_{PCU}(h_s - h_{PCU}) - \dot{M}_{bd}(h_s - h_{NLT51})}{Const}\right] - W_{PCU}$$

$\dot{M}_{NLP51}$ = RWCU regenerated heat exchanger inlet flow

4 Pennsylvania Power and Light Company Calculation EC-031-1008, 'Reactor Heat Balance for the Calculation of Core Thermal Power,' December 1996.

h_{NLT51} = RWCU outlet enthalpy, Btu/lb$_m$

h_{NLT52} = RWCU regenerated heat exchanger inlet enthalpy, Btu/lb$_m$

$\dot{M}_{PCU}$ = RWCU pump seal purge flow, lb$_m$/hr

h_{PCU} = RWCU pump seal purge flow enthalpy, Btu/lb$_m$

$\dot{M}_{bd}$ = RWCU Blowdown flow rate, lb$_m$/hr

W_{PCU} = Shaft work to RWCU pump, MW

Q_{PP} = Power transferred to recirculation pump seal purge flow, MW

$$= \left[\frac{\dot{M}_p(h_s - h_{CRD})}{Const}\right] \times (2\,pumps)$$

$\dot{M}_p$ = Reactor recirculation pump seal purge flow per pump, lb$_m$/hr

Q_P = Power added to the reactor coolant by the recirculation pumps, MW

= $(BHP - Q_c) \times$ (Number of pumps running)

= $\{\zeta \times MWP - Q_c\} \times$ (Number of pumps running)

BHP = Pump brake horsepower converted to MW

Q_c = Heat removed from recirculation pump seals by RBCCW per pump, MW

MWP = Power added to recirculation pump motor, MW

ζ = Recirculation pump motor efficiency

$$Q_{SAMPLE} = \frac{\dot{M}_{RWCUs}(h_s - h_{NLT51}) + \dot{M}_{FWs}(h_s - h_{fw}) + \dot{M}_{RS}(h_s - h_{RS})}{Const}$$

$\dot{M}_{RWCUs}$ = RWCU sample flow, lb$_m$/hr

$\dot{M}_{FWs}$ = Feedwater sample flow, lb$_m$/hr

h_{fw} = mass flow averaged feedwater enthalpy, Btu/lb$_m$

$$= \frac{\dot{M}_{FWA}h_{FWA} + \dot{M}_{FWB}h_{FWB} + \dot{M}_{FWC}h_{FWC}}{\{\dot{M}_{FWA} + \dot{M}_{FWB} + \dot{M}_{FWC}\}}$$

$\dot{M}_{RS}$ = Reactor recirculation pump sample flow rate, lb_m/hr

h_{RS} = Reactor recirculation pump sample enthalpy, Btu/lb_m

The values for Q_{RAD} and Q_{SAMPLE} are input constants in the calculation of total core thermal power with

$$Q_{RAD} = Q_{RAD}^{DW} + Q_{RAD}^{P} = (1.74 + 0.03)_{U1} and (1.55 + 0.03)_{U2} \quad \text{MW}$$

and $$Q_{SAMPLE} = 0.0 .$$

The values for the remainder of the terms in the calculation for total core thermal power are evaluated in real time based on the current values of the process variables. The total error in core flow measurement is composed of contributions from each of these terms. The error in each term may be random or systematic, and the effect on the total error is dependent upon whether the error is evaluated as random or systematic. Each of the terms is evaluated below.

Uncertainty Evaluation

All uncertainties and biases will be calculated, for both the current instrumentation and the proposed instrumentation upgrade, based on the guidance given in ASME-PTC-19.1 (1985)[5]. Based on this standard, the error, or uncertainty for any calculated variable, ***M***, which has the form

$$M = f(X_i)$$

where X_i are the measured variables that determine ***M*** and ***f*** is the functional form,

is determined as

$$dM = \sum_{i,j\neq i}^{n} \left.\frac{\partial f}{\partial X_i}\right|_{X_j} dX_i$$

where n is the total number of measured variables that go into determining the calculated variable, ***M***. The uncertainty determination, on a per unit basis is then

$$\frac{dM}{M} = \sum_{i,j\neq i} \left(\frac{X_i}{M} \left.\frac{\partial f}{\partial X_i}\right|_{X_j} \right) \frac{dX_i}{X_i}$$

where the terms in brackets are the sensitivity coefficients.

If the errors or biases in the individual elements (the dX_i/X_i terms in the above equation) are caused by the same boundary condition (for example, ambient temperature), the total error $\left(\frac{dM}{M}\right)$ is found by summing the appropriate terms in the above equation. If, as is more often the case, the errors or biases in the X_i are independent of each other, then the ASME recommends and probability theory[6] requires that the total uncertainty be determined by the square root of the sum of the squares of the terms as follows:

$$\frac{dM}{M}=\left[\sum_{i,j\neq i}\left(\left\{\frac{X_i}{M}\frac{\partial f}{\partial X_i}\bigg|_{X_j}\right\}\frac{dX}{X}\right)^2\right]^{1/2}$$

If the errors or biases in the individual elements are caused by a combination of boundary conditions, some independent and some systematic (that is, related) then a combination of the two procedures is used. Here M is a representation of the total core thermal power calculation. Based on the above discussion for the calculation for total core thermal power, and the fact that SSES has three feedwater flow loops where flow measurements are taken, the total core thermal power is

$$Q_{TP}=\sum_{i=1}^{3}\frac{\dot{M}_{fw,i}(h_s-h_{fw,i})}{Const}+Q_{RAD}+Q_{CRD}+Q_{CU}+Q_{PP}-Q_P-Q_{SAMPLE}$$

The radiation component, Q_{RAD}, is evaluated in Reference 4 to be 1.77 MW for Unit 1 and 1.58 MW for Unit 2. The component heat load from the control rod drive system, Q_{CRD}, is

$$Q_{CRD}=\frac{\dot{M}_{CRD}(h_s-h_{CRD})}{Const}=\frac{(32000)(1190.2-48)}{3.41214163\times10^6}=10.712\ \text{MW.}$$

The reactor water cleanup contribution to the overall heat balance is (Value for W_{PCU} = 0.0545 MW from Ref. 4)

$$Q_{CU}=\frac{(155{,}000)(525-413)+(1230)(1190.2-34.2)}{3.41214163\times10^6}-0.0545=5.5\ \text{MW.}$$

5 American Society of Mechanical Engineers Standard ASME-PTC-19.1 (1985).
6 See, for example, Moore, D. S. and McCabe, G. P. '**Introduction to the Practice of Statistics**,' W. H. Freeman and Co., New York, 1989.

Based on operating data, the power supplied to the reactor coolant by the recirculation pumps, Q_P, is approximately 7.5 MW. The power transferred to the recirculation pump seal purge flow, Q_{PP}, is

$$Q_{PP} = \left\{\frac{(1500-370)(1190.2-48)}{3.41214163\times10^6}\right\}(2\,pumps) = 0.76 \text{ MW.}$$

The energy contribution from the sampling system, Q_{SAMPLE}, is highly variable, according to Reference 4, and is taken as zero as a consequence. Therefore the total contribution of energy input to the reactor coolant, outside of the contribution from the feedwater system, is

$$Q_{RAD} + Q_{CRD} + Q_{CU} + Q_{PP} - Q_P - Q_{SAMPLE} = 1.77 + 10.71 + 5.5 + 0.76 - 7.5 - 0.0 = 11.2$$

MW for Unit 1 and 11.1 MW for Unit 2. This value is less than 0.33% of the total thermal heat load input to the reactor coolant. On that basis, the overall reactor heat balance can be written as

$$Q_{TP} = \sum_{i=1}^{3}\left\{\frac{\dot{M}_{fw,i}(h_s - h_{fw,i})}{Const}\right\} \pm Q_{LOSS}$$

where Q_{LOSS} represents the total gains and losses in the reactor coolant system other than accounted for by the feedwater flow balance. The error or uncertainty, dQ_{TP}, can then be evaluated as

$$dQ_{TP} = \frac{1}{Const}\left\{\left(\sum(h_s - h_{fw,i})d\dot{M}_{fw,i}\right) + \left(dh_s \sum \dot{M}_{fw,i}\right) - \left(\sum \dot{M}_{fw,i} dh_{fw,i}\right)\right\} \pm dQ_{LOSS}$$

Since the loss terms, dQ_{LOSS}, are to some extent dependent on some of the same variables as the feedwater flow, such as the enthalpies, these terms can be associated evenly between the three loops, so that, for an individual loop

$$dQ_{LOOP,i} = \frac{1}{Const}\left\{(h_s - h_{fw,i})d\dot{M}_{fw,i} + \dot{M}_{fw,i}(dh_s - dh_{fw,i})\right\} \pm \left(\tfrac{1}{3}\right)dQ_{LOSS}$$

The individual loop contribution to the overall heat balance, $Q_{LOOP,i}$, can be defined as

$$Q_{LOOP,i} = \frac{\dot{M}_{fw,i}(h_s - h_{fw,i})}{Const},$$

therefore, since the uncertainties in individual feedwater flow measurement loop are random

$$\frac{dQ_{TP}}{Q_{TP}} = \frac{1}{\sqrt{3}}\left\{\frac{d\dot{M}_{fw}}{\dot{M}_{fw}} + \frac{dh_s}{(h_s - h_{fw})} - \frac{dh_{fw}}{(h_s - h_{fw})}\right\} \pm \frac{dQ_{LOSS}}{Q_{LOSS}}$$

Each of these terms is evaluated individually in the presentation below, and the total error or uncertainty will be determined from a proper combination of these individual terms.

Feedwater flow

The algorithm for measuring feedwater flow at SSES, using the current venturi flow measurement devise is

$$\dot{M}_{fw,i} = K \cdot C_D \cdot \left(\frac{d^2}{\sqrt{1-\beta^4}} \right) \cdot F_A \cdot Y \cdot \sqrt{\rho \Delta p}$$

where K = dimensional constant converting from $(\rho\Delta P)^{1/2}(d)^2$ to mass flow

C_D = discharge coefficient for the nozzle

d = nozzle throat diameter, generally measured in inches

β = ratio of nozzle throat diameter to internal pipe diameter

Y = adiabatic expansion coefficient, which is unity for incompressible feedwater

ρ = fluid density, lb_m/ft^3

Δp = pressure differential between the throat and upstream of the nozzle

F_A = thermal expansion coefficient for the nozzle that accounts for thermal expansion of nozzle dimensions from ambient (calibration) temperature to feedwater flow measurement temperature. The dominant term in F_A is the nozzle throat dimension, *d*. Because the throat diameter dimension appears as a squared term in the algorithm, $F_A \cong 1 + 2\alpha(T - T_o)$, where α is the coefficient o linear expansion of the nozzle material.

T = feedwater temperature

and T_o = calibration temperature.

There are no uncertainties associated with ***K*** and ***Y***, since they are constants for incompressible fluids. Let

$$\Psi = \frac{C_D d^2}{\sqrt{1-\beta^4}}$$

where Ψ is determined by the nozzle calibration at a certified facility at a fluid temperature, T_o. Any biases in the nozzle diameter, ***d***, (which affects β) are imbedded in the calibration, and consequently, in Ψ. This parameter also embodies irregularities on nozzle shape and in piezometer taps normally accounted for in the discharge coefficient term, C_D. From the algorithm for uncertainty,

$$d\dot{M}_{fw} = \dot{M}_{fw}\left(\frac{d\Psi}{\Psi} + \frac{dF_A}{F_A} + \left(\frac{1}{2}\right)\left[\frac{d\rho}{\rho} + \frac{d\Delta p}{\Delta p}\right]\right).$$

For a compressed liquid, like feedwater $\rho = \rho(T_{fw}, p_{fw}) \cong \rho(T_{fw}, p_s)$. Therefore,

$$d\rho = \left.\frac{\partial\rho}{\partial T_{fw}}\right|_p dT_{fw} + \left.\frac{\partial\rho}{\partial p}\right|_{T_{fw}} dp + d\rho_{al}$$

where $d\rho_{al}$ accounts for the algorithm uncertainty in determining density from pressure and temperature.

The coefficient of thermal expansion, F_A, is a function of temperature, therefore,

$$dF_A = d(1 + 2\alpha(T_{fw} - T_o) = 2(T_{fw} - T_o)(d\alpha) + 2\alpha dT_{fw}$$

where the uncertainty in T_o is assumed to be zero and $d\alpha$ is the uncertainty in the thermal expansion properties of the nozzle material.

Since $F_A \cong 1.0, \frac{dF_A}{F_A} \cong 2(T_{fw} - T_o)(d\alpha) + 2\alpha(dT_{fw})$, the uncertainty in mass flow is

$$\frac{d\dot{M}_{fw}}{\dot{M}_{fw}} = \frac{d\Psi}{\Psi} + 2(T_{fw} - T_o)(d\alpha) + 2\alpha(dT_{fw}) + \left(\frac{1}{2\rho}\right)\left\{\left.\frac{\partial\rho}{\partial T_{fw}}\right|_p dT_{fw} + \left.\frac{\partial\rho}{\partial p}\right|_{T_{fw}} dp\right\} + \frac{d\rho_{al}}{2\rho} + \frac{d\Delta p}{2\Delta p}$$

Total Uncertainty in Power Level Calculation per Feedwater Loop

The total uncertainty in the power level calculation can be evaluated using the above relations and the total derivatives for steam and feedwater enthalpies. The feedwater conditions are subcooled; therefore feedwater enthalpy is a function of feedwater temperature and pressure. The steam conditions are saturated, and there is a small amount of moisture in the steam, in general, therefore, the steam enthalpy is a function of pressure and moisture content. Therefore,

$h_s = h_s(p,m)$ so that

$$dh_s = \left.\frac{\partial h_s}{\partial p}\right|_m dp + \left.\frac{\partial h_s}{\partial m}\right|_p dm + dh_{s,al}$$

where $m =$ the moisture content in the reactor exit steam, and $dh_{s,al}$ is the uncertainty in enthalpy determination resulting from uncertainty in the algorithm relating enthalpy to pressure.

The relationship for feedwater enthalpy is

$h_{fw} = h_{fw}(T_{fw}, p)$, thus

$$dh_{fw} = \left.\frac{\partial h_{fw}}{\partial T_{fw}}\right|_p dT_{fw} + \left.\frac{\partial h_{fw}}{\partial p}\right|_{T_{fw}} dp + dh_{fw,al}$$

where again the term $dh_{fw,al}$ is the uncertainty in calculating enthalpy from temperature and pressure.

The total uncertainty in the power level calculation, per feedwater measurement loop, is

$$\frac{dQ_{TP}}{Q_{TP}} = \frac{d\Psi}{\Psi} + 2(T_{fw} - T_o)(d\alpha) + \frac{1}{2}\frac{d\rho_{al}}{\rho} + \frac{1}{2}\frac{d(\Delta p)}{\Delta p} + \left(\frac{1}{(h_s - h_{fw})}\right)(dh_{s,al} - dh_{fw,al}) \pm \frac{dQ_{LOSS}}{Q_{LOSS}},$$

$$+\left(2\alpha + \frac{1}{2}\frac{1}{\rho}\left.\frac{\partial \rho}{\partial T}\right|_p - \frac{1}{(h_s - h_{fw})}\left.\frac{\partial h_{fw}}{\partial T_{fw}}\right|_p\right)dT_{fw} + \left\{\frac{1}{2}\frac{1}{\rho}\left.\frac{\partial \rho}{\partial p}\right|_{T_{fw}} + \left(\left.\frac{\partial h_s}{\partial p}\right|_m - \left.\frac{\partial h_{fw}}{\partial p}\right|_{T_{fw}}\right)\frac{1}{(h_s - h_{fw})}\right\}dp$$

$$+\frac{1}{(h_s - h_{fw})}\left.\frac{\partial h_s}{\partial m}\right|_p dm.$$

This equation embodies the sensitivity coefficients for the loop power terms that go into determining the total core thermal power. In general, it is assumed that, excepting the terms dependent on temperature and pressure, there are no systematic relationships for the terms in the equation. Therefore, the random error terms are combined as the square root of the sum of the squares.

B. After LEFM✓™ Feedwater Flow Element Installation

The core thermal power calculation is performed in a similar manner with the LEFM✓™ measurement system, with the exception that the measurement of feedwater flow is now determined by the LEFM✓™ technology. Based on a review of the Caldon Engineering Report (Reference 1), the algorithm for measuring feedwater flow is as follows, for an individual flow loop.

$$W_{loop} = (PF)\cdot(F_{A3})\cdot(r_p)\left(\sum_{i=1}^{4}\frac{w_i L_{ff,i}^2 \Delta t_i}{\left[(\tan\phi_i)\cdot(t_i + \left(\Delta t_i/2\right) - \tau_i)^2\right]}\right)$$

where W_{loop} = feedwater volumetric flow rate per measured loop, ft³/sec,

PF = profile factor, adapting the numerical integration of the LEFM✓™ to the specifics of the velocity profile,

F_{A3} = thermal expansion factor for the spool piece ID and $L_{ff,i}$,

r_p = internal radius of the piping, ft,

w_i = Gaussian integration weighting factor for path i; $w_1 = w_4 \approx 0.174$; $w_2 = w_3 \approx 0.326$,

$L_{ff,i}$ = transducer face-to –face distance in feet, measured between the wetted surfaces of the transducer wells,

ϕ_i = angle between path i and a normal to the spool piece flow axis; nominally, ϕ_i=45°,

t_i = total transit time of an acoustic pulse traveling in the direction of flow, along path i, seconds

Δt_i = difference in total transit times of pulses traveling against and with the flow along path i, seconds

and τ_i = that part of t_i which the pulse spends in non-fluid media, seconds.

The total mass flow for the feedwater system, $\dot{M}_{fw}$, is given by $\dot{M}_{fw} = \rho(p,T_{fw})\cdot\sum W_{fw,i}$, where $\rho(p.T_{fw})$ is the density of the feedwater, a function of feedwater temperature and pressure, since the feedwater is in the single phase liquid state, and $\sum W_{fw,i}$ is the sum of the three flow measurements in the three feedwater flow lines. The feedwater temperature is given by a semi-empirical correlation of ultrasonic velocity in the fluid and feedwater pressure. The mean ultrasonic velocity in the feedwater is given by the weighted average of the ultrasonic velocities measured for each acoustic path, thus,

$$c_{fw} = F_{A1}\sum_{i=1}^{4}\left[\frac{w_i \cdot L_{ff,i}}{\left(t_i - \tau_i + \left\{\Delta t_i/2\right\}\right)}\right]$$

where c_{fw} is the mean ultrasonic velocity in the feedwater system at rest and

F_{A1} is the thermal expansion factor for path length $L_{ff,i}$, a function of feedwater temperature only.

In the evaluation of current SSES uncertainty above, it was found that the total uncertainty was defined as

$$\frac{dQ_{TP}}{Q_{TP}} = \frac{d\dot{M}_{fw}}{\dot{M}_{fw}} + \frac{dh_s}{(h_s - h_{fw})} - \frac{dh_{fw}}{(h_s - h_{fw})} \pm \frac{dQ_{LOSS}}{Q_{LOSS}}$$

Since the only change made here is the change in the measurement of feedwater flow, the last three terms on the right hand side are not changed. Therefore, the uncertainty in feedwater flow measurement requires reevaluation. From the revised definition of feedwater mass flow,

$$\frac{d\dot{M}_{fw}}{\dot{M}_{fw}} = \frac{d\rho(p, T_{fw}(c_{fw}, p))}{\rho(p, T_{fw}(c_{fw}, p))} + \frac{dW_{fw}}{W_{fw}}$$

Based on the definition of feedwater volumetric flow rate,

$$\frac{dW_{fw}}{W_{fw}} = \frac{d(PF)}{PF} + \frac{dF_{A3}}{F_{A3}} + \frac{\sum_{i=1}^{4}\left\{\frac{w_i L_{ff,i}^2}{\tan\phi_i}\right\}\cdot d\left\{\frac{\Delta t_i}{\left(t_i - \tau_i + \left(\Delta t_i/2\right)\right)^2}\right\}}{\sum_{i=1}^{4}\frac{\left(w_i L_{ff,i}^2 \Delta t_i\right)}{\left(\tan\phi_i \cdot \left(t_i - \tau_i + \left(\Delta t_i/2\right)\right)^2\right)}}$$

where the biases in the geometric items, w_i, $L_{ff,i}$, r_p, and ϕ_i are imbedded in the measured profile factor. Let $t_{fi} = t_i - \tau_i + \Delta t_i/2$ and $\Delta t_i = t_{iu} - t_i$, where t_{iu} is the total time of flight opposite to the direction of flow. With these definitions and a fair amount of algebraic manipulation,

$$d\left[\frac{\Delta t_i}{\left(t_i - \tau_i + (\Delta t_i/2)\right)^2}\right] = \left(\frac{1}{t_{fi}^2}\right)\cdot\left[\left(1 - \left(\frac{\Delta t_i}{t_{fi}}\right)\right)\cdot dt_{iu} - \left(1 + \left(\frac{\Delta t_i}{t_{fi}}\right)\right)\cdot dt_i + 2\left(\frac{\Delta t_i}{t_{fi}}\right)\cdot d\tau_i\right].$$

Based on Appendix C in Reference 1, $t_{fi} = L_{ff,i}/c_i$, where c_i is the sound velocity along path i. Note that the nominal value of ϕ_i is 45°, thus $tan(\phi_i)$ is unity. In addition, analysis presented in Reference 1 shows that Δt_i is at least two orders of magnitude less than t_{fi}. Therefore,

$$(1+(\Delta t_i/t_{fi}))\approx(1-(\Delta t_i/t_{fi}))\approx 1.$$

Therefore,

$$\frac{dW_{fw}}{W_{fw}}\cong\frac{d(PF)}{PF}+\frac{dF_{A3}}{F_{A3}}+\frac{\sum_{i=1}^{4}(w_i\cdot c_i^2)\cdot\{dt_{iu}-dt_i+[2\Delta t_i/t_{fi}]\cdot d\tau_i\}}{\sum_{i=1}^{4}(w_i\cdot c_i^2\cdot\Delta t_i)}$$

Expressing each of the terms individually, noting that the feedwater temperature is nearly uniform through the measurement section (that is, $c_1=c_2=c_3=c_4=c$), noting that the Gaussian weighting factors, w_1 and w_4 are equal (thus $w_1 = w_4 = w_s$), and w_2 and w_3 are equal (that is, $w_2 = w_3 = w_l$), assuming that the path Δt's are the same (that is, $\Delta t_1\cong\Delta t_4\equiv\Delta t_s$ (subscript s for short path) and $\Delta t_2\cong\Delta t_3\equiv\Delta t_l$ (subscript l for long path)) and assuming that the short and long path uncertainties combine randomly, the above equation can be rewritten as

$$\frac{dW_{fw}}{W_{fw}}=\frac{d(PF)}{PF}+\frac{d(F_{A3})}{F_{A3}}+\left(\frac{\sqrt{2}}{2}\right)\cdot\left(\frac{(w_s)\cdot(dt_{su}-dt_s+(2\Delta t_s/t_{fs})\cdot d\tau_s)+(w_l)\cdot(dt_{lu}-dt_l+(2\Delta t_l/t_{fl})\cdot d\tau_l)}{(w_s\Delta t_s+w_l\Delta t_l)}\right)$$

Since the value of Δt is small compared with t_s, t_l, t_{su} and t_{lu}, the order of magnitude of the uncertainties are the same, thus, $dt_s\approx dt_l\approx dt_{su}\approx dt_{lu}$, and these uncertainties are uncorrelated, that is, random in nature. In addition, the uncertainties in τ are also the same order of magnitude. Probability theory (Reference 6) requires that random uncertainties be combined using the square root of the sum of the squares (that is, variations are additive, standard deviations are not) and also that the squared uncertainty terms be added, regardless of the sign of the term itself. Therefore, the terms in the above equation in t and τ can be combined, resulting in the following:

$$\frac{dW_{fw}(t,\tau)}{W_{fw}(t,\tau)}=\left\{\frac{\sqrt{2}}{2(w_s\Delta t_s+w_l\Delta t_l}\right\}\cdot\left[2w_s^2dt^2+2w_l^2dt^2+\left\{(2\frac{\Delta t_s}{t_{fs}}w_s)^2+(2\frac{\Delta t_l}{t_{fl}}w_l)^2\right\}\cdot d\tau^2\right]^{0.5}$$

and

$$\frac{dW_{fw}}{W_{fw}}=\frac{d(FA)}{FA}+\frac{d(F_{A3})}{F_{A3}}+\frac{dW_{fw}(t,\tau)}{W_{fw}(t,\tau)}.$$

The density and feedwater enthalpy used in the overall equation for the calculation of thermal power are functions of feedwater temperature and pressure. Since the feedwater temperature in the LEFM✓™ algorithm is determined from a correlation with the measured fluid speed of sound, the density and feedwater enthalpy are also functions of the speed of sound also. Thus,

$$\rho = \rho(p, T_{fw}(c)) \rightarrow d\rho = \left\{ \left.\frac{\partial \rho}{\partial T_{fw}}\right|_p \left.\frac{\partial T_{fw}}{\partial c}\right|_p dc + \left.\frac{\partial \rho}{\partial p}\right|_c dp + d\rho_{corr} \right\} \text{ and}$$

$$h_{fw} = h_{fw}(p, T_{fw}(c)) \rightarrow dh_{fw} = \left\{ \left.\frac{\partial h_{fw}}{\partial T_{fw}}\right|_p \left.\frac{\partial T_{fw}}{\partial c}\right|_p dc + \left.\frac{\partial h_{fw}}{\partial p}\right|_c dp + dh_{fw,corr} \right\}.$$

where $d\rho_{corr}$ includes the uncertainty in density due to the bias in the density/ temperature correlation **and** the uncertainty in density due to the bias in the temperature/sound velocity correlation

and $dh_{fw,corr}$ includes the uncertainty in enthalpy due to the bias in the enthalpy/temperature correlation **and** the uncertainty in the temperature due to the bias in the temperature/sound velocity correlation

Note that T_{fw} is a function only of speed of sound, *c*, here. Reference 1 states that T_{fw} is a function of speed of sound and pressure, but the reliance on pressure is very weak. Therefore, the feedwater temperature is evaluated in this study as a function of speed of sound only. Using these relations, the relationship for total core thermal power uncertainty may be rewritten as

$$\frac{dQ_{TP}}{Q_{TP}} = \frac{1}{\rho}\left\{ \left.\frac{\partial \rho}{\partial T_{fw}}\right|_p \left.\frac{\partial T_{fw}}{\partial c}\right|_p dc + \left.\frac{\partial \rho}{\partial p}\right| dp + d\rho_{corr} \right\} + \frac{dW_{fw}}{W_{fw}} + \frac{1}{(h_s - h_{fw})}\left\{ \left.\frac{\partial h_s}{\partial p}\right|_m dp + \left.\frac{\partial h_s}{\partial m}\right|_p dm \right\}$$

$$- \frac{1}{(h_s - h_{fw})}\left\{ \left.\frac{\partial h_{fw}}{\partial T_{fw}}\right|_p \left.\frac{\partial T_{fw}}{\partial c}\right|_p dc + \left.\frac{\partial h_{fw}}{\partial p}\right|_c dp + dh_{fw,corr} \right\} + \frac{dQ_{LOSS}}{Q_{LOSS}}.$$

The relationship for uncertainty in total calculated core thermal power depends, in part, on the uncertainty in the measured sound speed, c_{fw} in the LEFM✓™ algorithm, as shown above. Therefore, a relationship for the uncertainty in sound speed measurement must be developed. Taking the total derivative of the sound speed relationship results in

$$\frac{dc_{fw}}{c_{fw}} = \frac{d(F_{A1})}{F_{A1}} + \frac{\sum_{i=1}^{4}\left[\frac{w_i dL_{ff,i}}{t_{fi}} - \frac{w_i L_{ff,i}}{t_{fi}}\cdot\frac{dt_{fi}}{t_{fi}}\right]}{\sum_{i=1}^{4}\left(\frac{w_i L_{ff,i}}{t_{fi}}\right)}.$$

According to Reference 1, the correction term, F_{A1}, is approximately 1.003, therefore,

$\sum_{i=1}^{4}\left(\frac{w_i L_{ff,i}}{t_{fi}}\right) \cong c_{fw}\sum_{i=1}^{4} w_i = c_{fw}$, since $\frac{L_{ff,i}}{t_{fi}} \cong c_{fw}$ for each path, since the speed of sound in the feedwater fluid, c_{fw}, is nearly uniform across the pipe section, since the temperature is constant.

Therefore $$\frac{dc_{fw}}{c_{fw}} = \frac{d(F_{A1})}{F_{A1}} + \sum_{i=1}^{4}\left\{w_i\left(\frac{dL_{ff,i}}{L_{ff,i}} - \frac{dt_{fi}}{t_{fi}}\right)\right\}$$

From the previous discussion of time measurement uncertainty,

$$dt_{fi} = \left(\frac{1}{2}\right)\cdot(dt_{iu} + dt_i) - d\tau_i$$

The uncertainty in times, t_{iu} and t_i, are on the order of 40 nanoseconds and the uncertainty in the non-fluid delay time, τ, is on the order of 350 nanoseconds, according to Reference 1. Since the errors are not correlated, $dt_{fi} \cong -d\tau_i$.

Therefore,

$$\frac{dc_{fw}}{c_{fw}} = \frac{d(F_{A1})}{F_{A1}} + \sum_{i=1}^{4}\left(w_i\cdot\left[\frac{dL_{ff,i}}{L_{ff,i}} + \frac{d\tau_i}{t_{fi}}\right]\right).$$

The F_{A1} term corrects for the thermal expansion in the $L_{ff,i}$ paths, thus $F_{A1} = 1 + \alpha_e(T_{fw} - T_o)$, where α_e is the net thermal expansion coefficient for the $L_{ff,i}$. Therefore,

$$\frac{dF_{A1}}{F_{A1}} = \frac{(T_{fw} - T_o)\cdot d\alpha_e + \alpha_e\cdot dT_{fw}}{(1 + \alpha_e(T_{fw} - T_o))}.$$

If it is assumed that the length and time errors are uncorrelated, the uncertainty in sound speed becomes

$$\frac{dc_{fw}}{c_{fw}} = \frac{(T_{fw} - T_o)\cdot d\alpha_e + \alpha_e\cdot dT_{fw}}{(1 + \alpha_e(T_{fw} - T_o))} + \sqrt{2}w_s\left[\frac{dL_{ff,s}}{L_{ff,s}} + \frac{d\tau}{t_{fs}}\right] + \sqrt{2}w_l\left[\frac{dL_{ff,l}}{L_{ff,l}} + \frac{d\tau}{t_{fl}}\right]$$

If the assumptions are made that $\frac{dL_{ff,s}}{L_{ff,s}} \approx \frac{dL_{ff,l}}{L_{ff,l}}$ and $d\tau_s \approx d\tau_l$, the general expression for the total core thermal power uncertainty can be written as

$$\frac{dQ_{TP}}{Q_{TP}} = \left[\frac{1}{\rho}\frac{\partial\rho}{\partial p}\bigg|_c + \frac{1}{(h_s - h_{fw})}\left\{\frac{\partial h_s}{\partial p}\bigg|_m - \frac{\partial h_{fw}}{\partial p}\bigg|_c\right\}\right]dp + \frac{d(PF)}{PF} + \frac{\partial h_s}{\partial m}\bigg|_p \frac{dm}{(h_s - h_{fw})} + \frac{dQ_{LOSS}}{Q_{LOSS}} + \frac{d\rho_{corr}}{\rho} + \frac{dh_{fw,corr}}{(h_s - h_{fw})} +$$

$$\left[\sqrt{2}\cdot c_{fw}\left(\left\{\frac{w_s}{t_{fs}}\right\}^2 + \left\{\frac{w_l}{t_{fl}}\right\}^2\right)^{0.5} \cdot \left\{\frac{1}{\rho}\frac{\partial\rho}{\partial T_{fw}}\bigg|_p \cdot \frac{\partial T_{fw}}{\partial c}\bigg|_p - \frac{1}{(h_s - h_{fw})}\frac{\partial h_{fw}}{\partial T_{fw}}\bigg|_p \cdot \frac{\partial T_{fw}}{\partial c}\bigg|_p\right\}\right]\cdot d\tau +$$

$$\sqrt{2}\left[w_s^2 + w_l^2\right]^{0.5} \cdot \left\{\frac{c}{\rho}\cdot\frac{\partial\rho}{\partial T_{fw}}\bigg|_p \cdot \frac{\partial T_{fw}}{\partial c}\bigg|_p - \frac{c}{(h_s - h_{fw})}\cdot\frac{\partial h_{fw}}{\partial T_{fw}}\bigg|_p \cdot \frac{\partial T_{fw}}{\partial c}\bigg|_p\right\}\frac{dL_{ff,l}}{L_{ff,l}} + \frac{d(F_{A3})}{F_{A3}} +$$

$$\left\{\frac{\left(T_{fw} - T_o\right)\cdot d\alpha_e + \alpha_e \cdot dT_{fw}}{\left[1 + \alpha_e \cdot \left(T_{fw} - T_o\right)\right]}\right\}\cdot\left\{\frac{c}{\rho}\cdot\frac{\partial\rho}{\partial T_{fw}}\bigg|_p \cdot \frac{\partial T_{fw}}{\partial c}\bigg|_p - \frac{c}{(h_s - h_{fw})}\cdot\frac{\partial h_{fw}}{\partial T_{fw}}\bigg|_p \cdot \frac{\partial T_{fw}}{\partial c}\bigg|_p\right\} +$$

$$\left(\frac{\sqrt{2}}{2}\right)\cdot\left[\frac{\left(2dt^2\left[w_s^2 + w_l^2\right] + 4d\tau^2\left[\left(\frac{w_s\Delta t_s}{t_{fs}}\right)^2 + \left(\frac{w_l\Delta t_l}{t_{fl}}\right)^2\right]\right)^{0.5}}{\left(w_s\Delta t_s + w_l\Delta t_l\right)}\right]$$

RESULTS

a. Evaluation of Individual Terms – Current Feedwater Flow Measurement Configuration

Feedwater Flow Nozzle Coefficient

General Electric[7] specifies a feedwater flow element accuracy of ± 0.5% at the 95% confidence level after installation. The ASME[8] recommends that an uncertainty value of ± 0.25% be added algebraically to the overall uncertainty of the meter to account for differences between the calibration facility conditions and the conditions of

[7] General Electric Nuclear Energy Division Specification 22A1367AW.

the actual application. It is not clear whether the facility uncertainty is included in the flow element specification supplied by General Electric (Reference 7). Therefore, for purposes of this calculation, the overall accuracy of the individual flow element will be taken as ± 0.75%. Since the SSES design consists of three elements in parallel, and the uncertainty between flow elements is assumed random. The overall value of

$$\frac{d\Psi}{\Psi} = \frac{\pm 0.75\%}{\sqrt{3}} = 0.433\%.$$

Feedwater Flow Nozzle Thermal Expansion/Material Properties

An assumed uncertainty for the coefficient of thermal expansion for the stainless steel nozzle is ± 10%, and the coefficient of linear thermal expansion for stainless steel[9], $\alpha \cong 10\times10^{-6}\, {}^{in}\!/_{(in)(\circ F)}$, therefore, the total uncertainty due to materials expansion (assuming a 68°F calibration temperature and a feedwater temperature of 389°F) is

$$2(T_{fw} - T_o)(d\alpha) = 2(389-68)(10^{-6}) = 6.4\times10^{-4} = 0.06\%.$$

The bias in material properties is assumed to apply in like manner to both nozzles, since the nozzles were fabricated at the same time. Therefore, the total uncertainty contribution is 0.06% for the three loops combined.

Feedwater Flow Nozzle Thermal Expansion/Temperature

The term for nozzle thermal expansion uncertainties due to temperature measurement is $(2\alpha)(dT_{fw})$. The uncertainty in the final feedwater temperature measurement is ≈ ± 2.°F[10]. Therefore, the uncertainty due to feedwater temperature measurement affecting nozzle expansion is $\approx 2(10\times10^{-6})(2) = 4.0\times10^{-5}$ or 0.004%.

Feedwater Flow Nozzle Differential Pressure

The feedwater flow nozzle differential pressure transmitter at SSES is a Rosemont type 1151 Instrument. This type of flow transmitter has a typical uncertainty of ± 0.25% of full-scale reading. The uncertainty figure for the differential flow transmitter must include allowances for:

- transmitter biases due to pressurization (transmitters are calibrated at atmospheric pressure) including systematic uncertainties in the manufacturer-supplied pressure corrections

American Society of Mechanical Engineers, '**Fluid Meters,**' Sixth Edition, 1971.

9 Raznjevic, K. "**Handbook of Thermodynamic Tables and Charts**' Hemisphere Publishing Corporation, New York, N, 1976.

10 PPL Corporation Calculation EC-045-1003, November, 1995.

- systematic and random errors in calibration equipment
- drift in the analog to digital converters that provide the differential pressure signal to the plant computer (PICSY), which performs the feedwater flow and power computations.

The feedwater flow rate under current design conditions is $\approx 4.72\times10^6$ lb_m/hr[11]. The feedwater flow element and transmitter have a maximum value of 6×10^6 lb_m/hr, therefore the ratio of rated to full range pressure drop across the flow element is

$$\frac{\Delta p_{rated}}{\Delta p_{full}} = \left(\frac{\dot{M}_{rated}}{\dot{M}_{full}}\right)^2 = \left(\frac{4.72\times10^6}{6\times10^6}\right)^2 = 0.62$$

The uncertainty in nozzle differential pressure transmitter of $\approx \pm$ 0.25% accounts for the first two items of the list discussed above. The third item, relating to drift in the analog to digital conversion, is not known, but can be assumed to add $\leq$ 0.75% systematic error to the overall pressure transmitter error. Therefore the total error for the differential transmitter is

$$\left(\frac{1}{2}\right)\frac{d(\Delta p)}{\Delta p} = \left(\frac{1}{2}\right)\times\left(\frac{1}{\dfrac{\Delta p_{rated}}{\Delta p_{full}}}\right)\times\frac{1\%error}{\Delta p_{full}} = \left(\frac{1}{2}\right)\times\left(\frac{1}{0.62}\right)\times(1\%) = 0.81\%$$

Based on standard practice, it will be assumed that half of this uncertainty is systematic (errors in the ***dp's*** of each loop of this kind are likely to be of the same sign and same order) and half random. On this basis

$$\left(\frac{d\Delta p}{\Delta p}\right)_{sys} = \frac{(0.81)}{\sqrt{2}} = 0.57\%$$

$$\left(\frac{d\Delta p}{\Delta p}\right)_{random} = \frac{0.57}{\sqrt{3}} = 0.33\%$$

Feedwater Density/Correlation

The density correlation, determining feedwater density from the temperature and pressure measurement is carried out by the PICSY plant computer system. The accuracy of the density fit is assumed to be $\pm$ 0.1%, and is

11 Pankratz, D. R. and Faynshtein, K., 'Power Uprate Engineering Report for Susquehanna Steam Electric Station Units 1 and 2,' GE Nuclear Energy, NEDC-32161P, Class III (Proprietary), December 1993.

systematic, that is, all three feedwater measurement loops are affected in the same manner. The current conditions for the feedwater system are[12]:

$$\textit{Feedwater Pressure} = 1140\ psia$$

$$\textit{Feedwater Temperature} = 384\,°F$$

$$\textit{Feedwater Density}, \rho^{13} = \rho(1140\ psia, 384\,°F) = 54.60\ lb_m/ft^3.$$

Therefore,

$$\frac{1}{2}\frac{d\rho}{\rho} = \frac{(0.001)\cdot(54.60)}{2\cdot(54.60)} = 0.05\%$$

The total uncertainty associated with feedwater flow measurement is then

$$\frac{d\dot{M}_{fw}}{\dot{M}_{fw}} = \left\{ \left(\frac{d\Psi}{\Psi}\right)^2 + \left(\frac{dF_A}{F_A}\right)^2 + \left(\frac{1}{2}\frac{d\rho}{\rho}\right)^2 + \left(\frac{1}{2}\frac{d(\Delta p)}{\Delta p}\right)^2_{sys} + \left(\frac{1}{2}\frac{d(\Delta p)}{\Delta p}\right)^2_{random} \right\}^{1/2}$$

$$= \{ (0.433)^2 + (0.06)^2 + (0.004)^2 + (0.05)^2 + (0.57)^2 + (0.33)^2 \}^{0.5}$$

$$= 0.792\%$$

b. Evaluation of Other Uncertainties

Feedwater Density/Temperature

The uncertainty term for feedwater density with respect to temperature is given as $\left\{ \frac{1}{2}\frac{1}{\rho}\left.\frac{\partial\rho}{\partial T_{fw}}\right|_p \right\} dT_{fw}$. The assumed uncertainty in feedwater temperature is ± 2.°F. The term $\left.\frac{\partial\rho}{\partial T_{fw}}\right|_p$ is estimated from the ASME steam tables (Reference 11). Thus

$$\left.\frac{\partial\rho}{\partial T_{fw}}\right|_{p=1140} \approx \frac{\rho(390F,1140) - \rho(380F,1140)}{(390-380)} = \frac{54.360 - 54.753}{10} = -0.0393\ \text{lb}_m/(\text{ft}^3\ °\text{F}).$$

Since the assumed error in temperature measurement is $\Delta T_{fw} = \pm\, 2.0°F$, the overall error term is then

[12] These conditions are based on current measured plant parameters from Unit 2, May 5, 2000.

[13] American Society of Mechanical Engineers, '**ASME Steam Tables**,' Sixth Edition, 1987.

$$\frac{1}{2}\frac{1}{\rho}\frac{\partial \rho}{\partial T_{fw}}\bigg|_p dT_{fw} = \frac{1}{2}\frac{1}{\{54.60\}}\left\{\frac{(-0.0393)(2.0)}{\sqrt{3}}\right\} = \pm 0.0004 = \pm 0.04\% .$$

Feedwater Density/Pressure

The feedwater density/pressure term is given as

$$\left\{\frac{1}{2}\frac{1}{\rho}\frac{\partial \rho}{\partial p}\bigg|_{T_{fw}} dp\right\}.$$

From the ASME Steam Tables,

$$\frac{\partial \rho}{\partial p}\bigg|_{T_{fw}} = \frac{\rho(1200,387.5) - \rho(1100,387.5)}{100} = \frac{(54.4784)-(54.4468)}{100} = 3.16\times 10^{-4}\,\frac{lb_m}{(ft^3)(psi)}$$

The error in measuring pressure is ~± 0.02 of full range or ± (0.02)(1000) = 20 psi. Therefore, the uncertainty term for calculating density from pressure, for a three loop measurement system, is

$$\left\{\frac{1}{2}\frac{1}{\rho}\frac{\partial \rho}{\partial p}\bigg|_{T_{fw}}\right\}dp = \left\{\frac{1}{2}\frac{1}{(54.60)}\frac{(3.16\times 10^{-4})}{\sqrt{3}}\right\}(20) = 3\times 10^{-5} = 0.003\% .$$

Feedwater Enthalpy/Temperature

The relation for feedwater enthalpy uncertainty with respect to temperature is

$$\left\{\frac{1}{(h_s - h_{fw})}\frac{\partial h_{fw}}{\partial T_{fw}}\bigg|_p\right\}(dT_{fw}).$$

The feedwater temperature and pressure are 387.5°F and 1140 psia. The steam pressure in the reactor steam dome is 1045 psia. Thus,

$$(h_s - h_{fw}) = (h_{s,1045} - h_{fw}(1140,387.5)) = (1191.26 - 362.80) = 828.46\,{}^{Btu}\!/_{lb_m},$$

and $$\frac{\partial h_{fw}}{\partial T_{fw}}\bigg|_p \approx \frac{(h_{fw}(1140,390) - h_{fw}(1140,380))}{(10)} = \frac{(365.46)-(354.85)}{(10)} = 1.061\left\{{}^{Btu}\!/_{(lb_m)(F)}\right\}.$$

The temperature uncertainty is ± 2.0°F, therefore,

$$\left\{\frac{1}{(h_s-h_{fw})}\left.\frac{\partial h_{fw}}{\partial T_{fw}}\right|_p\right\}dT_{fw}=\left\{\frac{1}{(828.46)}\frac{(1.061)}{\sqrt{3}}\right\}(2.0)=0.00148=0.148\%.$$

Feedwater Enthalpy/Pressure

The term for the uncertainty in enthalpy due to pressure uncertainty is given as

$$\left\{-\frac{1}{(h_s-h_{fw})}\left.\frac{\partial h_{fw}}{\partial p}\right|_{T_{fw}}\right\}dp.$$

From the ASME Steam Tables,

$$\left.\frac{\partial h_{fw}}{\partial p}\right|_{T_{fw}}\approx\frac{h_{fw}(1200,387.5)-h_{fw}((1100,387.5)}{100}=\frac{(362.879)-(362.748)}{100}=1.31\times10^{-3}\,Btu/(lb_m)(psi)$$

Since the enthalpy difference and the pressure measurement uncertainty are the same,

$$\left\{-\frac{1}{(h_s-h_{fw})}\left.\frac{\partial h_{fw}}{\partial p}\right|_{T_{fw}}\right\}dp=\left\{-\frac{1}{(828.46)}\frac{(1.31\times10^{-3})}{\sqrt{3}}\right\}(20)=-1.8\times10^{-5}=-0.0018\%.$$

Steam Enthalpy/Moisture

The steam produced in the reactor at SSES leaves the reactor vessel with a moisture content of ~ 0.4%. From thermodynamics relationships,[14] the steam quality, x, is defined as the mass ratio of steam to total fluid in a steam water mixture, that is,

$$x=\frac{SteamMass}{TotalMass}=\frac{h_s-h_g}{h_g-h_f}$$

where h_s = steam enthalpy leaving the reactor vessel, Btu/lb$_m$

h_g = enthalpy of saturated steam, Btu/lb$_m$

and h_f = enthalpy of saturated liquid, Btu/lb$_m$.

Typically, the saturated steam and liquid enthalpies are functions only of reactor steam pressure. Therefore,

$$h_s=(1-x)\cdot h_f(p)+x\cdot h_g(p).$$

The moisture content, $m=(1-x)$, thus

$$h_s = m \cdot h_f(p) + (1-m) \cdot h_g(p), \text{ or}$$

$$h_s = h_g(p) - m \cdot (h_g(p) - h_f(p)) = h_g(p) - m \cdot (h_{fg}(p)).$$

Therefore, $$\left.\frac{\partial h_s}{\partial m}\right|_p = -h_{fg}(p).$$

The moisture content at SSES is designed to be ~ 0.4% and does not exceed 0.5%, therefore, the uncertainty in moisture content can be taken as between zero and 0.5%, or ± 0.25%. Therefore, the uncertainty in steam enthalpy due to moisture content is given by

$$\frac{1}{(h_s - h_{fw})}\left.\frac{\partial h_s}{\partial m}\right|_p dm = \frac{-h_{fg}(1045)}{(h_s - h_{fw})}(0.0025) = \frac{-(641.86)(0.0025)}{828.46} = 0.0019 \text{ or } 0.19\%.$$

This error is assumed systematic, since the steam measurements will be taken from essentially the same source. Therefore, there will be no reduction due to the increased number of measurements.

Steam Enthalpy/Pressure

The uncertainty in steam enthalpy due to pressure variation is given by

$$\frac{1}{(h_s - h_{fw})}\left.\frac{\partial h_s}{\partial p}\right|_m dp.$$

From the ASME Steam Tables,

$$\left.\frac{\partial h_s}{\partial p}\right|_m \approx \frac{h_g(1100) - h_g(1000)}{(1100-1000)} = \frac{(1189.08)-(1192.95)}{(100)} = -3.87 \times 10^{-2}\ Btu/\{(lb_m)(psi)\}$$

Therefore, $$\frac{1}{(h_s - h_{fw})}\left.\frac{\partial h_s}{\partial p}\right|_m dp = \frac{(-3.87\times10^{-2})(20)}{(828.46)} = -0.0009 = -0.09\%$$

Again, this error is assumed systematic, and is unchanged by the increased number of measurements.

Other Gains and Losses

The other gains and losses in the determination of total reactor thermal power are:

a. Thermal radiation losses from the reactor pressure vessel and piping, ≈ ± 1.7 MW

b. Losses from the control rod drive system, ≈ ± 10.7 MW

14 See, for example, Haberman, W. L. and John, J. E. A., '**Engineering Thermodynamics**,' Allyn and Bacon, Inc., Boston, MA, 1980.

c. Losses to the reactor water cleanup system, ≈ ± 5.6 MW

d. Losses to the recirculation pump seal flow, ≈ ± 0.76 MW

e. Gains from the recirculation pump heat, ≈ ± 7.5 MW

and f. Losses to the reactor sampling system, ≈ ± 0.00 MW.

Since the terms are all very small, using a conservative ten percent uncertainty for each of the terms and combining them as the root sum of the squares will result in a conservative estimate of the total uncertainty from these terms. Thus

$$dQ_{LOSS} = \{(0.17)^2 + (1.07)^2 + (0.56)^2 + (0.076)^2 + (0.75)^2 + (0.0)^2\}^{0.5}$$

$$dQ_{LOSS} = 1.434 \text{ MW}.$$

Thus $$\frac{dQ_{LOSS}}{Q_{TP}} = \frac{1.434}{3441} = 0.00042 = 0.042\%$$

c. Total Core Thermal Power Measurement Uncertainty.

As discussed above, the total core thermal power measurement uncertainty for the currently installed instrumentation is the square root of the sum of the squares of the individual uncertainties. Since each of the individual terms, and manufacturers data, gives 95% confidence intervals, the end result is at least a 95% confidence interval estimate. The overall uncertainty is then

$$\frac{dQ_{TP}}{Q_{TP}} = \left\{\left(\frac{1}{3}\right)\sum\left(\frac{d\dot{M}_{fw}}{\dot{M}_{fw}}\right)^2 + \left(\frac{dQ_{LOSS}}{Q_{TP}}\right)^2\right\}^{0.5}$$

where the Q_{loss} terms are the other terms evaluated. Thus

$$\frac{dQ_{TP}}{Q_{TP}} = \{0.792^2 + 0.04^2 + 0.003^2 + 0.148^2 + 0.0018^2 + 0.19^2 + 0.09^2 + 0.042^2\}^{0.5}$$

$$\frac{dQ_{TP}}{Q_{TP}} = 0.835\%$$

d. Evaluation of Individual Terms – Following Implementation of the LEFM✓™ System

Thermal Expansion Factor for the Individual Path Lengths

The thermal expansion factor for the path lengths, F_{Al}, is given as $F_{Al} = 1 + \alpha_e(T_{fw} - T_o)$. Therefore,

$$\frac{dF_{A1}}{F_{A1}} = \frac{\left(T_{fw} - T_o\right) \cdot d\alpha_e + \alpha_e \cdot dT_{fw}}{1 + \alpha_e\left(T_{fw} - T_o\right)}.$$

Based on information in Reference 1, the following values are used:

$\alpha_e \approx 7 \times 10^{-6}$

$dT_{fw} \sim 1°$ F (max).

Therefore, $$\frac{\alpha_e \cdot dT_{fw}}{1 + \alpha_e\left(T_{fw} - T_o\right)} \approx \frac{\left(7 \times 10^{-6}\right) \cdot (1.0)}{\left[1 + \left(7 \times 10^{-6}\right) \cdot \{390 - 100\}\right]} \approx 7 \times 10^{-6} = 0.0007\%$$

Using a value for $d\alpha_e = (0.1) \cdot \alpha_e$, as is typical in ASME stress analysis[15], it is found that

$$\frac{\left(T_{fw} - T_o\right) \cdot d\alpha_e}{\left(1 + \alpha_e \cdot \left(T_{fw} - T_o\right)\right)} = \frac{(390 - 100) \cdot (0.1) \cdot \left(7 \times 10^{-6}\right)}{\left[1 + \left(7 \times 10^{-6}\right) \cdot (390 - 100)\right]} = \frac{2.03 \times 10^{-4}}{1.00203} = 2 \times 10^{-4} = 0.02\%.$$

Therefore, $$\frac{dF_{A1}}{F_{A1}} = \sqrt{(0.0002)^2 + \left(7 \times 10^{-6}\right)^2} = 0.0002 = 0.02\%.$$

Thermal Expansion Factor for the Spool Piece and Transit Length

The thermal expansion factor for the spool piece and transit length, $F_{A3} = 1 + 3\alpha_e \cdot \left(T_{fw} - T_o\right)$. Therefore,

$$\frac{dF_{A3}}{F_{A3}} = \frac{3 \cdot \left(\left(T_{fw} - T_o\right) \cdot d\alpha_e + \alpha_e \cdot dT_{fw}\right)}{\left[1 + 3 \cdot \alpha_e\left(T_{fw} - T_o\right)\right]}.$$

Using the same values for the parameters as above,

$$\frac{3 \cdot \alpha_e \cdot dT_{fw}}{\left[1 + 3 \cdot \alpha_e \cdot \left(T_{fw} - T_o\right)\right]} = \frac{3 \cdot \left(7 \times 10^{-6}\right) \cdot (1.0)}{\left[1 + \left(3 \cdot 7 \times 10^{-6}\right) \cdot (390 - 100)\right]} = \frac{\left(2.1 \times 10^{-5}\right)}{(1.006)} = 2.1 \times 10^{-5} = 0.002\%,$$

and $$\frac{3 \cdot \left(T_{fw} - T_o\right) \cdot d\alpha_e}{\left[1 + 3 \cdot \alpha_e \cdot \left(T_{fw} - T_o\right)\right]} = \frac{3 \cdot (390 - 100) \cdot (0.1) \cdot \left(7 \times 10^{-6}\right)}{(1.006)} = 0.00061 = 0.061\%.$$

Therefore, $$\frac{dF_{A3}}{F_{A3}} = \sqrt{\left(2 \times 10^{-5}\right)^2 + \left(6.1 \times 10^{-4}\right)^2} = 0.00061 = 0.061\%.$$

Profile Factor

Based on information in Reference 1, the allocation of uncertainties for the profile factor is:

[15] Ibid., Reference 1, Appendix E.

Facility	±0.25%	based on a standard certified test at Alden Research Labs.
Test LEFM✓™	±0.15%	See Reference 1, Appendix E, for basis details.
Measurement Uncertainty Modeling and Reynolds No.	±0.25%	See Reference 1, Appendix E, for discussion
Observational Uncertainty	±0.10%	See Reference 1, Appendix E, for discussion.

Since these uncertainties are all random, the total uncertainty due to the profile factor is

$$\frac{d(PF)}{PF} = \sqrt{(0.25)^2 + (0.15)^2 + (0.25)^2 + (0.10)^2} = \sqrt{(0.1575)} = 0.40\%$$

Spool Piece Thermal Expansion, Material Properties and Temperature

The terms for spool piece thermal expansion, material properties and temperature uncertainties are given as

$$\sqrt{\left(\frac{d(F_{A3})}{F_{A3}}\right)^2 + \left(\left\{\frac{(T_{fw}-T_o)\cdot d\alpha_e + \alpha_e \cdot dT_{fw}}{[1+\alpha_e\cdot(T_{fw}-T_o)]}\right\}\cdot\left\{\left.\frac{\partial T_{fw}}{\partial c}\right|_p\cdot\left[\left(\frac{c}{\rho}\right)\cdot\left.\frac{\partial \rho}{\partial T_{fw}}\right|_p - \left(\frac{c}{(h_s-h_{fw})}\right)\cdot\left.\frac{\partial h_{fw}}{\partial T_{fw}}\right|_p\right]\right\}\right)^2}$$

Evaluating terms:

$$\frac{d(F_{A3})}{F_{A3}} = 0.061\%$$

$$\left[\frac{(T_{fw}-T_o)\cdot d\alpha_e + \alpha_e\cdot dT_{fw}}{[1+\alpha_e\cdot(T_{fw}-T_o)]}\right] = 0.02\%$$

$$\left.\frac{\partial T_{fw}}{\partial c}\right|_p = \frac{3}{16}\frac{(F)}{(ft/\sec)}$$ See Reference 1, Appendix C

$$\left.\frac{\partial \rho}{\partial T_{fw}}\right|_p \cong -0.0393\frac{lb_m}{(ft^3)\cdot(F)}$$ From Section b, above

$$c \approx 4200\, ft/\sec$$

$$\rho = 54.60\, lb_m/ft^3$$

$$\left.\frac{\partial h_{fw}}{\partial T_{fw}}\right|_p = 1.061\frac{Btu}{(lb_m)\cdot(F)}$$

$$\left(h_s - h_{fw}\right) = 828.46\, Btu/lb_m$$

Then, combining terms, it is found that

$$\sqrt{0.061^2 + \left((0.02)\cdot\left[\left\{\frac{3}{16}\right\}\cdot\left(\left[\frac{4200}{54.60}\right]\cdot(-0.0393) - \left[\frac{4200}{828.46}\right]\cdot(1.061)\right)\right]\right)^2} = \sqrt{0.061^2 + 0.032^2} = 0.069\%$$

Time of Flight Measurement

This term is evaluated in Appendix E of Reference 1 and includes the total measurement uncertainty resulting from the pulse transit times. The total uncertainty is evaluated as ±0.18% for a single flow measurement and is the uncertainty is evaluated as random, therefore, the total time of flight uncertainty for three measurements combined is

$$\frac{\pm 0.18}{\sqrt{3}} = \pm 0.104\%$$

Non-Fluid Delay Uncertainty

The no-fluid delay uncertainty is evaluated in Appendix E of Reference 1, and is composed of uncertainties in the amount of time the ultrasonic pulses are outside to the fluid. The total uncertainty for a single flow element is evaluated as ±0.094%. The uncertainty is again evaluated as random, therefore, for three loops, the error is

$$\frac{\pm 0.094}{\sqrt{3}} = \pm 0.054\%.$$

Subtotal: Volumetric Flow Measurement

Based on the above information, the total uncertainty in volumetric flow measurement is

$$\frac{dW_{fw}}{W_{fw}} = \left[\left(\frac{dW_{fw}}{W_{fw}}\right)_{PF}^2 + \left(\frac{dW_{fw}}{W_{fw}}\right)_{\dim}^2 + \left(\frac{dW_{fw}}{W_{fw}}\right)_{align}^2 + \left(\frac{dW_{fw}}{W_{fw}}\right)_{X}^2 + \left(\frac{dW_{fw}}{W_{fw}}\right)_{TF}^2 + \left(\frac{dW_{fw}}{W_{fw}}\right)_{NF}^2\right]^{0.5}$$

$$= \left[0.4^2 + 0.1^2 + 0.058^2 + 0.069^2 + 0.104^2 + 0.054^2\right]^{0.5} = 0.438\%.$$

Note: The terms $\left(\frac{dW_{fw}}{W_{fw}}\right)_{\dim}$ and $\left(\frac{dW_{fw}}{W_{fw}}\right)_{align}$ are terms allocated for the spool piece dimensions and alignment, respectively, and are evaluated, in detail, in Reference 1, Appendix E.

This value is plant specific for the three-loop SSES feedwater configuration.

Feedwater Density/Correlation

The term $\left(\frac{d\rho_{corr}}{\rho}\right)$ is evaluated based on information supplied in Reference, which states that $d\rho_{corr}$ is accurate to within ± 0.02 lb_m/ft^3. Therefore, $\frac{d\rho_{corr}}{\rho} = \frac{(0.02)}{(54.412)} = 0.00037 \approx 0.04\%$.

Feedwater Density/Temperature Uncertainty

The uncertainty term relating the density/temperature relationship is given as

$\frac{1}{\rho}\cdot\left.\frac{\partial\rho}{\partial T_{fw}}\right|_p \cdot dT_{fw}$, where the $\boldsymbol{dT_{fw}}$ term comprises the total uncertainty associated with measuring feedwater temperature. This term is made up of the uncertainty in spool piece dimensions affecting temperature measurement, $\left[2w_s^2 + 2w_l^2\right]^{0.5}\cdot\left[c\cdot\left.\frac{\partial T_{fw}}{\partial c}\right|_p\right]\cdot\left(\frac{dL_{ff}}{L_{ff}}\right)$, the temperature/time of flight/non-fluid delay effect on temperature measurement, $\left[2\left(\frac{w_s}{t_{fs}}\right)^2 + 2\left(\frac{w_l}{t_{fl}}\right)^2\right]^{0.5}\cdot\left.\frac{\partial T_{fw}}{\partial c}\right|_p \cdot c\cdot d\tau$, and two terms relating to the temperature uncertainty with respect to pressure (± 0.05°F) and the overall temperature correlation (± 0.5°F). The total temperature uncertainty is then evaluated to be less than 1°F, but the 1°F value is used for conservatism.

Therefore, since

$$\left.\frac{\partial\rho}{\partial T_{fw}}\right|_p = -0.0393\left(\frac{lb_m}{ft^3\cdot F}\right)$$

$$\rho = 54.412\frac{lb_m}{ft^3}$$

and $$dT_{fw} \le 1F,$$

then $$\frac{1}{\rho}\cdot\left.\frac{\partial\rho}{\partial T_{fw}}\right|_p \cdot dT_{fw} = \left(-\frac{(0.0393)}{(54.412)}\right)\cdot(1) = 0.0007 = 0.07\%.$$

It is assumed that half of this uncertainty is systematic and half is random, therefore, the total uncertainty for three measurement loops is

$$\sqrt{\left(\frac{0.07^2}{2}\right)+\frac{1}{3}\left(\frac{0.07^2}{2}\right)}=0.06\%.$$

Feedwater Density/Pressure Uncertainty

The density pressure uncertainty term is given by

$$\frac{1}{\rho}\cdot\left.\frac{\partial\rho}{\partial p}\right|_{T_{fw}}\cdot dp.$$

Evaluating individual terms, as in section b above.

$$\left.\frac{\partial\rho}{\partial p}\right|_{T_{fw}}=3.16\times10^{-4}\frac{lb_m}{(ft^3)(psi)}$$

and $dp = 50 psi$, thus

$$\frac{1}{\rho}\cdot\left.\frac{\partial\rho}{\partial p}\right|_{T_{fw}}\cdot dp=\frac{(3.16\times10^{-4})\cdot(50)}{(54.412)}=2.9\times10^{-4}\approx0.03\%.$$ Since pressure is input, any uncertainty is likely to be systematic, therefore, there is no reduction in uncertainty for multiple loops.

Feedwater Enthalpy/Temperature Uncertainty

The value of this term is the same as for the case prior to the implementation of the LEFM✓™ system, with the exception that the dT_{fw} is reduced from ± 2.0°F to ± 1°F. Thus

$$\frac{1}{(h_s-h_{fw})}\cdot\left.\frac{\partial h_{fw}}{\partial T_{fw}}\right|_p\cdot dT_{fw}=\frac{(1.061)}{(828.48)}\cdot(1)=0.00128=0.128\%.$$

This uncertainty is characterized as half systematic and half random. Therefore, for the three loop configuration

$$\sqrt{\left(\frac{0.128^2}{2}\right)+\frac{1}{3}\cdot\left(\frac{0.128^2}{2}\right)}=0.00105\approx0.11\%.$$

Feedwater Enthalpy/Pressure Uncertainty

This term has the same value as prior to the installation of the LEFM✓™ instrumentation or 0.0018%.

Steam Enthalpy/Moisture Uncertainty

The value for this term is unchanged as a result of the LEFM✓™ instrumentation installation or 0.19%

Steam Enthalpy/Pressure Uncertainty

The steam enthalpy/pressure uncertainty is given by

$$\left.\frac{\partial h_s}{\partial p}\right|_m \cdot \frac{dp}{(h_s - h_{fw})}.$$

From the previous calculation for this uncertainty,

$$\left.\frac{\partial h_s}{\partial p}\right|_m = -3.87\times10^{-2}\ Btu/\{(lb_m)(psi)\},$$

and $(h_s - h_{fw}) = 828.46\, Btu/lb_m$,

Using $dp = 50$ psi,

$$\left.\frac{\partial h_s}{\partial p}\right|_m \cdot \frac{dp}{(h_s - h_{fw})} = \frac{(-3.87\times10^{-2})\cdot(50)}{(828.46)} = -0.0023 = -0.23\%.$$

Again, since pressure is input, the uncertainty is likely to be systematic, therefore, the uncertainty is unchanged in going to three measurement loops.

Other Gains and Losses

This term is the same regardless of what type of feedwater flow measurement system is used, therefore the value for this uncertainty is 0.07%.

Total Uncertainty in Overall Core Thermal Power Calculation

The total 95% confidence level uncertainty is the square root of the sum of the squares of all items considered, thus

$$\frac{dQ_{TP}}{Q_{TP}} = \sqrt{(0.438^2)+(0.04^2)+(0.06^2)+(0.03^2)+(0.11^2)+(0.0018^2)+(0.19^2)+(0.23^2)+(0.07)^2}$$

$$= \sqrt{0.304} = 0.551\%.$$

References

1. 'TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System' CALDON, Incorporated Engineering Report 80P, Revision 0, March 1997.
2. United States Atomic Energy Commission, 'Power Levels of Nuclear Power Plants,' Rev. 1, December 1973.
3. NRC SER on Caldon Topical and TU Submittal.
4. Pennsylvania Power and Light Company Calculation EC-031-1008, 'Reactor Heat Balance for the Calculation of Core Thermal Power,' December 1996.
5. American Society of Mechanical Engineers Standard ASME-PTC-19.1 (1985).
6. Moore, D. S. and McCabe, G. P., '**Introduction to the Practice of Statistics**,' W. H. Freeman and Co., New York, 1989.
7. General Electric Nuclear Energy Division Specification 22A1367AW.
8. American Society of Mechanical Engineers, '**Fluid Meters**,' Sixth Edition, 1971.
9. Raznjevic, K. "**Handbook of Thermodynamic Tables and Charts**' Hemisphere Publishing Corporation, New York, N, 1976.
10. PPL Corporation Calculation, EC-045-1003, "Feedwater Flow Element Transmitter and Loop Calibration Guidelines," November, 1995.
11. Pankratz, D. R. and Faynshtein, K., 'Power Uprate Engineering Report for Susquehanna Steam Electric Station Units 1 and 2,' GE Nuclear Energy, NEDC-32161P, Class III (Proprietary), December 1993.
12. American Society of Mechanical Engineers, '**ASME Steam Tables**,' Sixth Edition, 1987.
13. Haberman, W. L. and John, J. E. A., '**Engineering Thermodynamics**,' Allyn and Bacon, Inc., Boston, MA, 1980.

Attachment 2 to PLA-5300
Power Ascension Testing

Section 10.7 of the SSES Power Uprate Licensing Topical Report NE-2000-001P Rev. 1 delineated a startup test plan. This plan as delineated therein requires revision. For convenience, Section 10.7 of SSES Power Uprate Licensing Topical Report NE-2000-001P Rev. 1 is provided below. Following this, is the revised test plan. The basis for the changes are provided in the revised test plan.

Section 10.7 of the SSES Power Uprate Licensing Topical Report NE-2000-001P Rev. 1 stated the following:

10.7 SUMMARY OF STARTUP TESTS

Compared to the initial startup test program, operation at the increased licensed core thermal power level requires only limited startup tests. The testing for increased core thermal power operation will be conducted in accordance with Reference 1.6. The following tests will be performed to assure adequate performance at the increased core thermal power conditions:

1. **Testing will be performed on any instrumentation that requires recalibration for increased core thermal power operation.**

2. **Steady-state data will be taken at points from 90% up to the previous rated thermal power, so that operating performance parameters can be projected for increased power operation before the previous licensed power level is exceeded. The LEFM✓™ will be the measurement source of record for these tests, with the previously used venturi flow meter serving a backup.**

3. **Power increases beyond the previous licensed power level will be made along an established flow control/rod line in at least one intermediary step between the previous licensed power level and the new licensed power level. Steady-state operating data will be taken and evaluated at each step, using the newly installed LEFM✓™ system as the source of record for feedwater flow measurement and core thermal power calculation. Since the licensed power increase is small compared with the previous power uprate, the two step process is deemed appropriate.**

4. **Control system tests will be performed for the feedwater reactor water level controls and pressure controls. These operational checks will be made at the previous licensed thermal power condition and at the new licensed thermal power condition to show acceptable adjustment and operational capacity. The small increase in licensed core thermal power makes performing the adjustments only at the end points reasonable.**

The recommended startup test approach is a series of small increases in steam flow, roughly 0.5% steam flow increase per step, followed by a series of Test Procedures (TP's) completed at each plateau. The Test Procedures will examine turbine and feedwater control stability and assure that the plant is operating as expected. The test program will proceed, and the individual Test Procedures will be reviewed and approved according to established plant procedures

The revised Section 10.7 delineating the Startup test process to be implemented is provided below. The reasons for the changes are identified in the italicized text following each changed statement(s).

The revised startup test plan is as follows:

10.7 SUMMARY OF STARTUP TESTS

Compared to the initial startup test program, operation at the increased licensed core thermal power level requires only limited startup tests. The testing for increased core thermal power operation will be conducted in accordance with Reference 1.6. The following tests will be performed to assure adequate performance at the increased core thermal power conditions:

1. **Testing will be performed on any instrumentation that requires recalibration for increased core thermal power operation.**

2. **Steady-state data will be taken at points from 90% up to the previous rated thermal power, so that operating performance parameters can be projected for increased power operation before the previous licensed power level is exceeded. The LEFM✓™ will be the measurement source of record for these tests, with the previously used venturi flow meter serving a backup.**

3. **Power increases beyond the previous licensed power level will be made along an established flow control/rod line in at least one intermediary step between the previous licensed power level and the new licensed power level. Steady-state operating data will be taken and evaluated at each step, using the newly installed LEFM✓™ system as the source of record for feedwater flow measurement and core thermal power calculation. Since the licensed power increase is small compared with the previous power uprate, the two step process is deemed appropriate.**

4. **The reactor pressure control system will be monitored up to and including the new rated thermal power to ensure that control system deadband is small enough to limit steady state limit cycles (if any) to a reasonable amount. Power increases beyond the previous licensed power level will be made along an established flow control/rod line in at least one intermediary step between the previous licensed power level and the new licensed power level.** *[The reasons for not performing control system stability tests at 3441 MWt and higher power levels are numerous. It is not common practice to perform Feedwater and Pressure Regulator control system stability tests at such high power levels. The concern with performing such testing is not Nuclear Steam Supply System (NSSS) performance. The concern is that abrupt step changes in vessel water level or pressure regulator setpoints required by the testing will result in large Balance of Plant (BOP) system transients primarily due to the interaction of turbine control valves. This could result in unnecessary plant challenges to BOP systems with potential for reduction in plant power or for plant shutdown. Safety of the plant during such testing is not the concern.*

   Sufficient data was taken and analyzed during the 1994 power uprate test program to predict stable response of the feedwater and pressure regulator control systems at increased power levels up to and including the new RTP.

Power increases beyond the previous licensed power level will be made along an established flow control/rod line in at least one intermediary step between the previous licensed power level and the new licensed power level. [First sentence revised and second sentence deleted to be consistent with the changes made in item 4 above.] The test program will proceed, and the individual Test Procedures will be reviewed and approved according to established plant procedures.