U.S. NUCLEAR REGULATORY COMMISSION

REGION V

AUGMENTED INSPECTION TEAM (AIT) REPORT

Report No. 50-397/92-30

Docket No. 50-397

License No. NPF-21

Licensee: Washington Public Power Supply System (WPPSS)
Nuclear Plant, Unit 2
Richland, Washington

Inspection at: Washington Nuclear Plant, Unit 2
Benton County, Washington

Inspection Conducted: August 17-21 and 27-29, 1992

Team Members:
L. F. Miller Jr., Chief, Reactor Safety Branch, Region V, Team Leader
L. E. Phillips, Chief, Core Performance Section, Reactor Systems Branch, NRR
J. W. Clifford, Acting Chief, Program Development Section, Operator Licensing Branch, NRR
W. Ang, Senior Project Inspector, Reactor Projects Section 1, Region V
D. L. Proulx, Resident Inspector, WNP-2, Region V
T. Sundsmo, Project Inspector, Reactor Projects Section 2, Region V
Dr. J. March-Leuba, Consultant, Oak Ridge National Laboratory (ORNL)

Approved by:
K. E. Perkins, Jr., Director, Division of Reactor Safety and Projects, Region V

Inspection Summary:

Inspection on August 17-21 and 27-29, 1992 (Report No. 50-397/92-30)

Areas Inspected: Augmented Inspection Team (AIT) review of a power oscillation event at WNP-2 on August 15, 1992.

During this inspection, Inspection Procedure 93800 was used.

9210070002 920929
PDR ADOCK 05000397
G PDR

TABLE OF CONTENTS

ENCLOSURE 1

INSPECTION REPORT

50-397/92-30

REPORT DETAILS

1. Introduction - Management Summary and Event Overview

1.1 Purpose and Scope of the AIT Inspection

This report presents the findings of an NRC Augmented Inspection Team (AIT) inspection of the power oscillation event which occurred on August 15, 1992 at the Washington Nuclear Plant, Unit 2 (WNP-2) facility.

The decision to dispatch an AIT was made by NRC management based on the apparent similarity of this event to others which have occurred at boiling water reactors, most notably at the La Salle Unit 2 facility on March 9, 1988. Also, control of power oscillation was the subject of NRC Bulletin 88-07, "Power Oscillations in Boiling Water Reactors." The NRC staff and the Boiling Water Reactor Owners' Group (BWROG) have continued to study this phenomena since that event, to develop technical specifications and other guidance to avoid power oscillations.

The WNP-2 event was of particular concern because it was initiated from a region of power and flow which was outside the exclusion areas defined by the technical specifications, but which was also within the cautionary area suggested in a March 18, 1992 BWROG advisory letter, "Implementation Guidance for Stability Interim Corrective Actions."

The AIT consisted of six NRC inspectors or engineers, and a consultant from Oak Ridge National Laboratory. The consultant was expert in the field of power oscillations at BWRs. The AIT Charter (Appendix H) directed that the team verify the circumstances, identify the causes of the event, and evaluate its significance.

The inspection was conducted from August 17-21 and 27-29, 1992. An entrance meeting was held with the licensee on August 17, 1992 at WNP-2. A public exit meeting was held at the corporate headquarters on August 29, 1992. Appendix I provides a list of attendees at the exit meeting.

1.2 Inspection Methodology

After an initial briefing by licensee personnel at the entrance meeting on August 17, 1992, the AIT interviewed the operating crew and Shift Technical Advisor, engineers who participated in the fuel reload analysis, and licensee managers. The AIT reviewed the licensee's core physics information regarding the event, and made independent calculations for comparison. The AIT also reviewed other relevant charts, logs, written statements, procedures, memoranda, and other documentation during the inspection. Region V management was briefed daily on the progress and preliminary findings of the inspection.

1.3 Event Summary

At 0300:38, with the reactor at 34 % power and 32 % flow, on August 15, 1992, a manual scram of the reactor was initiated due to power oscillations observed by the control room crew. The oscillations began at 0258, and were observed by the operators at 0259. The oscillations in core average power were approximately 25 % power in amplitude peak-to-peak. Post-event analysis confirmed that the oscillations were in-phase, across the core. No fuel damage occurred, but the operating limit critical power ratio was exceeded. The reactor tripped normally, and all systems performed as expected. An unusual event was declared at 0320 due to the power oscillations, and terminated at 0430.

1.4 Findings and Conclusions

The AIT made numerous observations, findings, and conclusions which are detailed in this report. The following findings and conclusions are considered to be the most significant ones identified:

1.4.1 Findings

- There was no evidence of fuel failures or violation of fuel safety limits due to the power oscillations which occurred.

- The primary cause of the oscillations was very skewed radial and axial power distributions in the reactor. These were a result of: the control rod pattern selected for power escalation and recirculation pump shifting, and the core fuel loading configuration.

- The large radial and axial peaking factors obtained during the event exeeded the values assumed in current BWROG procedures for stability region boundaries. These peaking factors were, therefore, not conservative with respect to the less limiting, empirical stability regions specified in the Technical Specifications. The large peaking factors resulted in instability predictable by stability calculation codes (such as LAPUR).

1.4.2 Conclusions

- The current reactor can be operated with a very low risk of power oscillations by following the procedures and startup plan proposed by the licensee in their letter to the NRC dated August 29, 1992. This plan includes operation of the stability monitor when in the region greater than 25 % power and less than 50 % flow. It also required the specification of stable rod patterns to be used during power operations with less than 50 % flow.

- Neither the licensee nor the fuel vendor properly assessed the vulnerability of the reactor to instability when operated as permitted by the licensee's procedures, when they designed the Cycle 7 and Cycle 8 fuel reloads.

- The licensee did not adequately incorporate into its procedures the March 18, 1992 BWROG advisory letter that recommended increased instability alertness outside the TS exclusion regions.

- Operator training was not effective in ensuring operator understanding of the latest BWROG advisory letter. The training and qualification program for Shift Technical Advisor/Shift Nuclear

Engineers (STA/SNEs) did not adequately address the potential impact of power distributions and rod patterns on reactor stability.

- Procedural controls to specify appropriate control rod patterns or other effective stability criteria between 20 % power and the target full power rod pattern were inadequate.

2. Narrative Description of Event of August 15, 1992

On August 13, 1992 at 1655 PST, drywell unidentified leakage was identified. The licensee commenced a shutdown to identify the leak. Operators reduced power to 5 % to make an entry into the drywell to attempt to identify and isolate the leak. At 0746 on August 14, Supply System personnel entered the drywell, found a leaking valve, backseated it, and stopped the leak.

Subsequently, at 1710, the Reactor Operator (RO) commenced rod withdrawal to resume power operations. At 2109, with reactor at 15 % power, the main generator was synchronized to the grid. At 2228, the mid-shift crew relieved the watch. At 2320, the reactor was at 25 % power, and operators continued to increase power with control rods. The stability monitor (the Advanced Neutron Noise Analysis, or "ANNA" system) was not in operation.

The procedures used by the crew during the shift on which the event occurred are listed in Appendix K (Procedures 1-4). A summary of the shift staffing during the event is provided in Appendix J.

During the reactor startup, two operational constraints limited the reactor power and flow conditions where flux shaping and recirculation pump shifting from slow speed to fast speed could be performed. First, reactor feed flow of approximately 4.5 to 5 million pounds per hour and reactor power greater than 34 % was required by procedure PPM 3.1.2, "Reactor Cold Start-up," prior to the pump shift in order to prevent cavitation in the recirculation system. Second, the length of time that the recirculation pumps were operated in fast speed with less than 50 % flow needed to be minimized, because excessive pump vibration occurred under those conditions. To minimize this time period, control rod patterns were adjusted prior to the pump shifts to obtain a desirable flux shape and reduce the number of fully inserted control rods. These adjustments minimized the amount of rod movements after the pump shifts, and would have allowed efficient control rod withdrawal / power ascension within the constraints of the fuel preconditioning limits. Fuel preconditioning limits could have restricted control rod withdrawal if power density near the control rods exceeded specific limits. Usually, fully inserted control rods (i.e., less than position 12) are the most limiting. If preconditioning limits had been approached, full power operation would have been delayed while several power changes (via recirculation flow control) were performed to produce power and xenon distributions that permitted control rod withdrawal.

CRO#1, under supervision of the STA/SNE, was adjusting control rods from 1830 (August 14, 1992) until about 0245. This period was used to adjust the reactor's neutron flux profile, adjust the timing of six control rods, and conduct a surveillance to exercise a control rod. Reactor power and total core flow were maintained at about 36 % and 30 %, respectively, during these adjustments. The STA then informed the CRS and SM that the control rod adjustments were complete. CRO#1 closed the "A" recirculation loop flow control valve (FCV) in preparation for shifting the "A" recirculation pump to fast speed. As the FCV was closed, reactor power decreased from 36.4 % to 33.5 %, and total core flow decreased from 30.5 % to 26.0 %.

Reactor power oscillations started as the FCV was closing, at about 0258:18. Average Power Range Meter (APRM) oscillations were initially observed by the operators at about 0259:49, and were terminated by a manual scram that was activated at 0300:38. The SM, CRS, CRO#1, CRO#3, and the STA were present in the control room. CRO#1 initially identified the power oscillations when he observed the APRMs swinging about 20 % peak to peak power. CRO#1 alerted the CRS and other crew members. The crew then observed multiple Local Power Range Meter (LPRM) downscale indications, and continued power oscillations on the APRMs. The CRS recommended initiating a manual reactor scram to the SM, who then directed the scram. The manual scram effectively terminated the reactor power oscillations.

Following the manual scram, reactor water level went below Level 3 (13 inches) for about 30 seconds, causing the crew to momentarily enter emergency operating procedure (EOP) PPM 5.1.1, "RPV Control (Non-ATWS)." The lowest reactor water level reached was -15 inches; it was automatically restored by the main feedwater pumps without requiring operator action. CRO#2 was called into the control room (from the control room back panels) and assisted with the post scram actions. The crew continued plant shut down, without complications, using procedure PPM 3.3.1, "Scram Recovery."

Plant computer data showed that the power oscillations were core wide and were in phase. The AIT concluded that power oscillations lasted for 144 seconds from initial onset until the reactor scram. Operators scrammed the reactor 49 seconds following the first annunciator received (APRM Flow biased rod block). Licensee calculations indicated a Minimum Critical Power Ratio (MCPR) of 1.68 during the event. The safety limit (minimum allowed value) MCPR of 1.07 was not exceeded. A reactor coolant sample confirmed that no fuel damage had occurred.

All other plant systems responded as expected.

3. Description and Analysis of Power Oscillations which Occurred

This section describes the power oscillations that occurred on August 15, 1992, in the WNP-2 plant, and the analyses performed by the AIT to identify the root causes for these oscillations.

The subject of boiling water reactor thermal hydraulic stability has been of interest to designers, operators, and regulators since the early days of BWR design. Much theoretical, experimental, and operational information has accumulated on the subject. BWR stability is influenced by several power distribution and operating state variables that change during normal operation and from cycle to cycle. In particular, the radial and axial power distributions and core inlet subcooling have a strong impact on stability.

3.1 Description of the Oscillations

The August 15, 1992 WNP-2 power oscillations exhibited the characteristics of a density-wave instability of the corewide type (also called in-phase or fundamental mode). Of the four types of instabilities that have been observed in BWRs (corewide, out-of-phase, single channel, and control-system-induced instabilities), the corewide type of power oscillation is the type least likely to result in a significant challenge to the fuel because, under most reasonable operating conditions, the high APRM automatic scram will take effect before any thermal limits are violated.

Figure 1 shows a time trace for LPRM 32-17C during the event. On this trace, the oscillations started approximately at 02:58:45 and grew for about one minute with a decay ratio of approximately 1.06 until the amplitude saturated to a peak-to-peak value that is approximately 80 % of the average local power. The oscillation frequency was approximately 0.5 Hz (2 second period). The decay ratio (DR) is a measure of the relative stability of the reactor; DR values less than 1.0 indicate stable operation, while DRs greater than 1.0 indicate an instability.

3.1.1 Oscillation Amplitude

A characteristic of corewide type instabilities is that the oscillation amplitude is proportional to the average value of the local power at each core location (i.e., Local Power Range Monitor (LPRM) readings). In other words, for corewide oscillations, the oscillation amplitude should not be the same in all LPRM signals, but it should be proportional to the average LPRM reading. On first evaluation of the data, this appeared not to be the case because LPRMs 32-17C and 32-09A had an oscillation amplitude several times larger than all other signals recorded. Later detailed evaluations indicated that all signals (from the process computer) except LPRMs 32-17C and 32-09A had been conditioned by a 0.3 Hz low-pass filter that reduced the apparent oscillation amplitude by a factor of four. For instance, the filtered Average Power Range Monitor (APRM) data indicated an apparent oscillation peak-to-peak amplitude of only 6 % of core rated power, while other unfiltered recorded data (from the Transient Data Acquisition System (TDAS)) showed that the APRM had oscillated between 22.66 % and 48.91 % of nominal power, indicating that the APRM had a peak-to-peak oscillation of at least 26 % of core rated power.

Since the core average power during the event was 33.7 % of core rated power, the APRM peak-to-peak oscillation measured as a percent of the actual average power during the event was 77 % (i.e., 26/.337), which is consistent with the observed 80 % relative oscillations in all LPRMs once the effect of the 0.3 Hz low-pass filter was corrected.

3.1.2 Oscillation Frequency and Decay Ratio

From the APRM recorder the power oscillations started at 02:58:18 and grew for approximately one minute until they saturated. The estimated decay ratio during the oscillation growth period was 1.06, and the

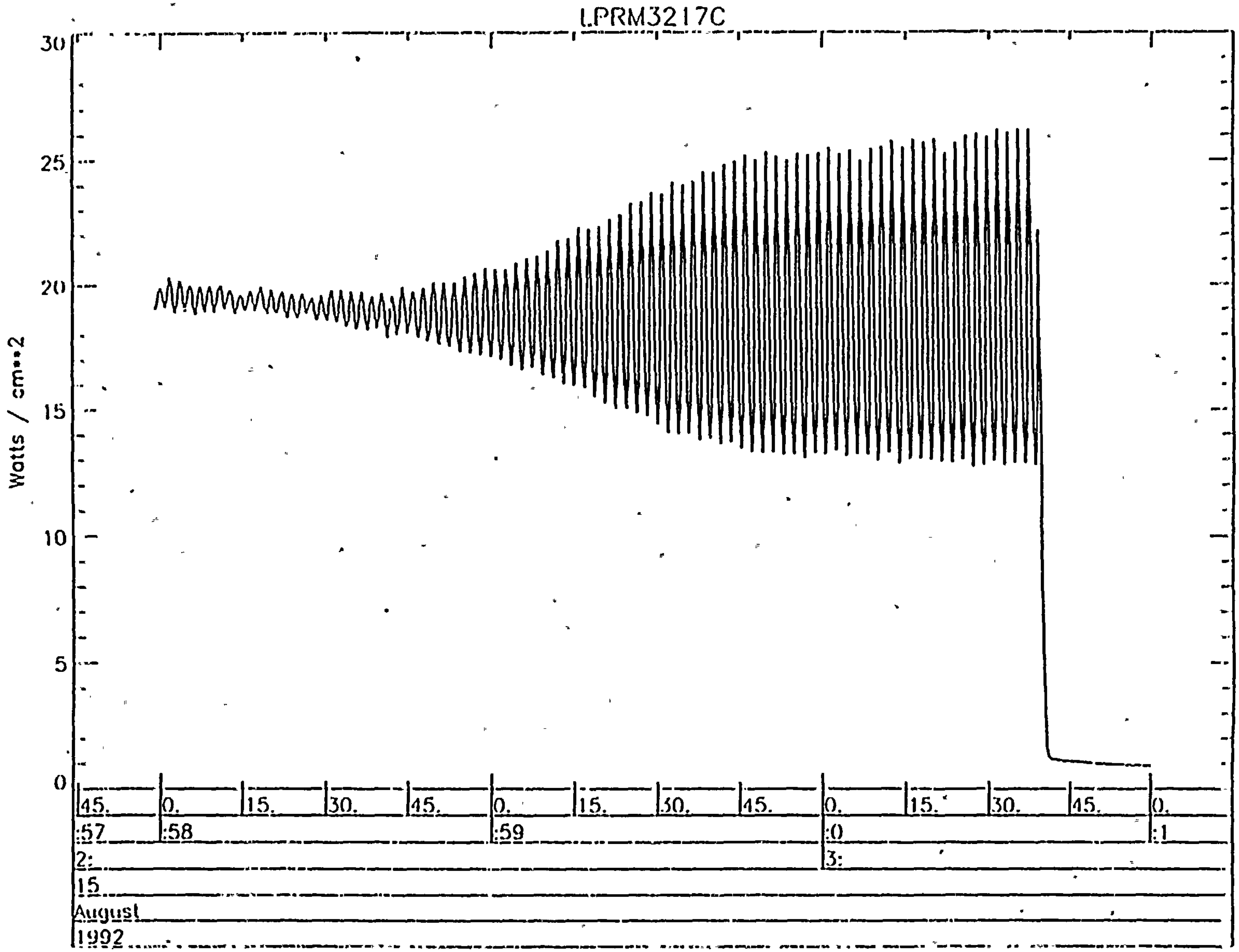

Figure 1. Time trace of LPRM 32-17-C during the 8/15 event. This LPRM was not filtered by the 0.3 Hz low-pass filter.

oscillation frequency was 0.5 Hz.[1] (Note that superscripts refer to Appendix A, "Notes.")

At 0300:00 , the oscillations seemed to have reached a stable limit cycle (i.e., decay ratio of 1.0), and their amplitude was not growing significantly. A slight trend towards increased amplitude, however, can be observed in Figure 1 even at the moment of scram. This small increase is attributed to nonstationary reactor conditions, such as nonequilibrium feedwater temperature and nonequilibrium xenon.

3.1.3 Oscillation's Effect on Fuel Integrity

There was no evidence of fuel failure or violation of fuel safety limits due to the August 15 power oscillations. However, the licensee's calculations showed that the thermal-margin operating limit (OLMCPR) was exceeded.

The change in critical power ratio (CPR) during the oscillations was evaluated by the licensee using the VIPRE transient code (under review by the NRC), and by Siemens Nuclear Power (SNP) Corporation using the XCOBRA-T code.[2] The results of both calculations indicate that the change in CPR was relatively small and not sufficient to cause a safety limit minimum CPR (SLMCPR) violation.

A calculation was performed by SNP using a bounding analyses according to the licensed methodology (COTRANSA 2). This analysis resulted in a CPR change of 0.27 and a MCPR of 1.68, well above the SLMCPR of 1.07, but exceeding the Operating Limit minimum CPR of 1.795.

The above results are consistent with previous experience and calculations, which indicated that safety limits were not likely to be violated for relatively small-amplitude power oscillations such as were observed at WNP-2.

3.2 Stability Calculations for the August 15 Event

Using the best estimates available for power, flow, power distribution, and the operating conditions at the time of the oscillations, the AIT calculated a corewide DR of 1.05, a hot-channel DR of 0.83, and an out-of-phase DR of 1.0.

These results indicate that:

(1) The hot channel was probably stable, but not with much margin (DR 0.83). A single channel thermohydraulic instability was not likely, but could not be ruled out without further analyses because of the extreme radial power peaking existing in this event.

(2) Even though the event data shows that the instability was clearly of the in-phase or fundamental mode, the calculations indicated that the out-of-phase mode was also fairly unstable. From these results, it was not certain which of the two modes would likely dominate. Therefore, other

startups with these skewed power distributions could result in out-of-phase oscillations.

Oak Ridge National Laboratory (ORNL) performed some sensitivity analyses for the AIT to identify the root cause of this instability. As stated before, the main root cause was the extreme radial and axial power distribution. The other contributing parameter was the mixed-core characteristics present in WNP-2 at the time of the event.
Appendices C, D, and E provide a more complete discussion of the stability calculations performed by ORNL for the AIT.

3.3 Causes for Instability which Occurred

The AIT concluded that the main cause of this instability event was the very skewed radial and axial power distribution (1.92 radial peaking factor and up to 1.76 axial peaking factor). This same core had been started on two previous occasions (July 27, 1992, and August 2, 1992) without oscillations, even though the recirculation pump upshift was performed at higher rod lines on the previous startups. The power distribution during the August 15 startup was caused by the control rod pattern selected, which included all shaper rods withdrawn and four primary power rods located in the core-center region that were withdrawn 28 notches, resulting in a high power area in the core-center region. When a more conservative control rod pattern is used for the pump upshift, the decay ratio for this core can be as low as 0.3 (see Appendix D), compared to a decay ratio of 1.05 with the rod pattern selected on August 15.

The AIT also found by analyses that a contributor to the instability of Cycle 8 in WNP-2 was a mixed core with unbalanced flow characteristics between the new SNP 9x9-9X fuel and the old 8x8 assemblies. Under these conditions, the low-power and low resistance 8x8 bundles were starving the flow from the high-power and high friction 9x9-9X bundles; this effect can be observed in Figure 2, which shows the relative power and flow of all the channels in the core at the time of the event. LAPUR calculations (Appendix D) indicated that if the whole core had been loaded with 8x8 fuel, the decay ratio would have been 20 % lower and the instability would have been avoided, even with the power distributions which were in use on August 15. Noticeably, if the whole core had been loaded with 9x9-9X instead of being a mixed core, the decay ratio would be lower by 10 % and the instability may have also been avoided.

AIT LAPUR calculations indicated that the hot channel was thermohydraulically stable during the August 15 event. AIT LAPUR calculations also indicated that the out-of-phase mode of instability did not have much margin to instability. If the LAPUR calculations had been performed before the event, they would have shown that the instability could have been either in-phase or out-of-phase with almost equal probability.

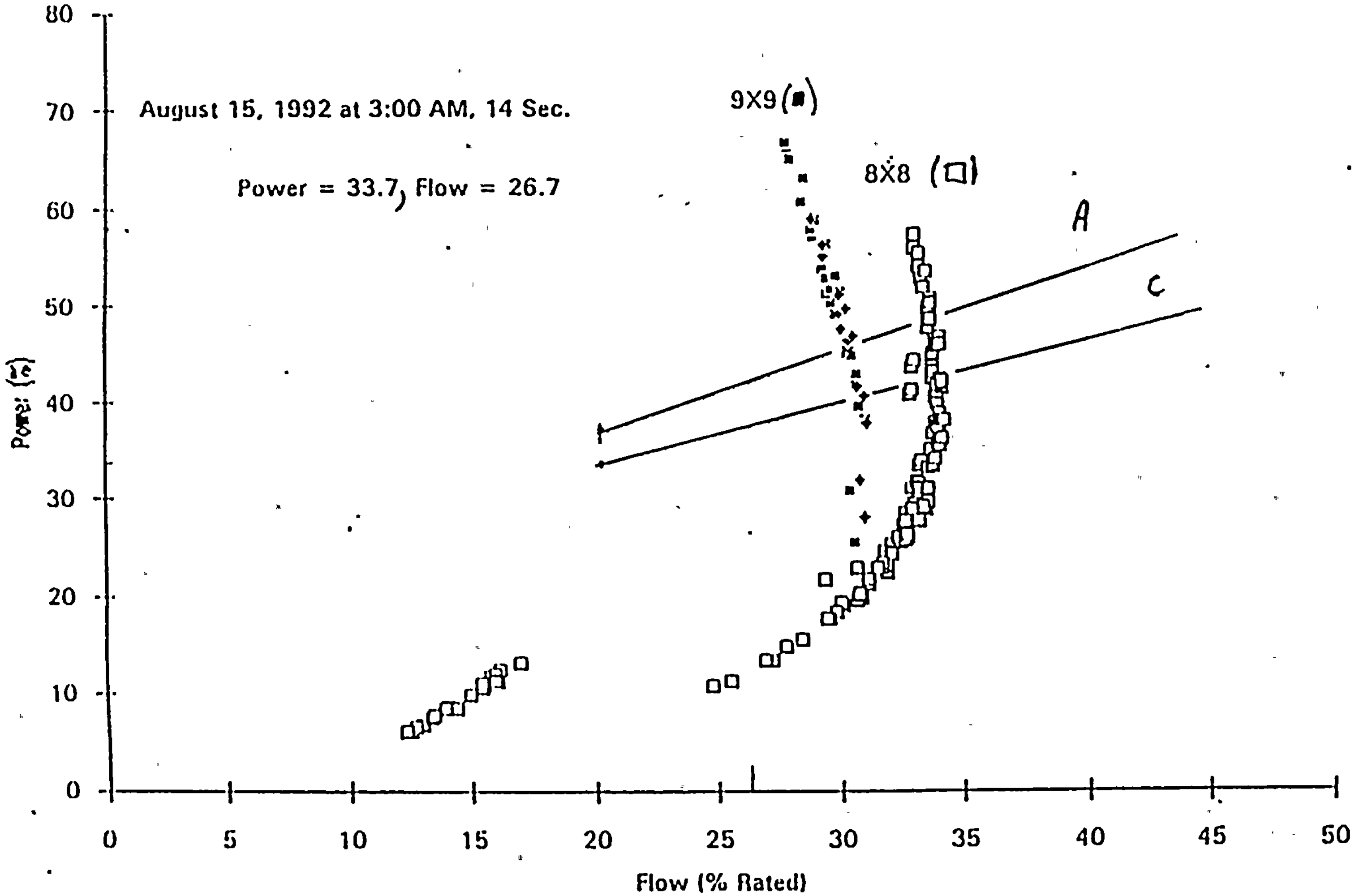

Figure 2. Relative power and flow of all channels just before the 8/15 WNP-2 event. The reduced flow area of the 9x9-9X fuel compared to 8x8 results in a lower flow.

4. Review of Responses to Generic Correspondence

This section of the report evaluates the licensee's responses to generic correspondence concerning core power oscillations.

4.1 Review of Generic Correspondence

Since the core power oscillation event at LaSalle in 1988, the NRC and BWROG have developed guidance intended to reduce the likelihood of a core power oscillation event, and to mitigate an event should core power oscillations occur.

The NRC issued NRC Bulletin (NRCB) 88-07 dated June 15, 1988 and NRCB 88-07 Supplement 1 dated December 30, 1988. The bulletin established actions required in response to the core power instability event at LaSalle, including operator and STA training on recognition, prevention, and mitigation of uncontrolled power oscillations. The bulletin also called for a verification of instrument adequacy.

The supplement provided additional actions. It stated that the specific power to flow boundaries for plants using non-GE supplied fuel should be based on existing boundaries that had been previously approved by the NRC. For new fuel designs, it stated that the stability boundaries should be reevaluated and justified based on calculated changes in core decay ratio using NRC approved methodology, decay ratio measurements, or applicable operating experience.

The BWROG has issued several relevant letters: BWROG-8847 dated July 8, 1988, BWROG-8879 dated November 3, 1988, a revision to BWROG-8879 dated November 4, 1988, and BWROG-92030 dated March 18, 1992. The first letter (BWROG-8847) documented the BWR Owners' Group understanding of a June 24, 1988 meeting with the NRC, including information to be developed by the BWROG to address the types of issues identified in NRC Bulletin 88-07. No specific actions were identified in this letter.

The second letter (BWROG-8879) provided interim recommendations, developed by GE to address potential instabilities in BWRs, that the BWROG considered prudent interim actions while the results of ongoing BWROG Stability Programs were evaluated. The BWROG requested an expedited review and decision on the part of all BWR Owners to the recommendations in the letter. The actions were grouped according to whether BWR 4 plants had filtered (Group 1) or unfiltered (Group 2) APRM signals. BWR 5 and 6 plants, including WNP-2, were grouped with the filtered APRM plants (Group 2). Group 2 plants were to immediately scram the reactor to exit Region A (greater than 100 % rod line with less than 40 % rated core flow). A third letter (BWROG-92030) modified the second letter (BWROG-8879) to make it clear that the plant operators were to scram the reactor if any thermal hydraulic instability occurred while in specified regions of the power to flow map.

The latest BWROG letter dated March 18, 1992 provided additional recommendations that the BWROG felt were necessary to enhance the effectiveness of the interim corrective actions (from BWROG-8879). The BWROG provided this information based on additional analyses and additional operating experience, including an oscillation event at a non-U.S. reactor that initiated outside the recognized instability region. The letter stated that "the guidance should be carefully considered by all owners" for application to their procedures and training programs. The recommendations also were based on an increased understanding of the conditions that might result in oscillations. Based on these considerations, the BWROG highlighted the need for caution when operating near the stability exclusion region.

The additional cautions included:

1) emphasizing operator training to scram the reactor even if oscillations were observed but were less than 10 %,

2) enhanced guidance for recognition of oscillations, including increase or periodicity in noise levels when near the instability region (defined as approximately 5 % power or 5 % flow from the current potential instability region),

3) a caution that the stability exclusion region boundaries were not exact, especially at lower power and flow where the uncertainties in measuring power and flow can increase, along with an admonition that it was best to minimize the amount of time spent operating near the stability exclusion region,

4) the importance of the effect of reduced feedwater temperature on core stability, and

5) the strong impact of radial and axial power distributions and core inlet subcooling on stability, as well as the concern that fuel and reload designs have resulted in higher fuel bundle power levels, making oscillations more likely.

4.2 Implementation of Generic Correspondence

The AIT reviewed the licensee's procedures referenced in this report, and correspondence listed in Items 6-9 of Appendix K, and conducted interviews to evaluate the licensee's implementation of generic information related to prevention and mitigation of core power oscillations.

Based on review of these documents, interviews with plant and management personnel, and evaluation of procedures and training, the AIT found that the licensee had implemented some of the information available on core power instabilities. The licensee's actions included the following:

- PPM 4.12.4.7, "Power Plant Maneuvering," was developed in response to NRCB 88-07 through the utility's formal procedure review process. This procedure was subsequently modified to incorporate the guidance of the BWROG November 3, 1988 letter (as modified by the November 4, 1988 BWROG letter) and NRCB 88-07 Supplement 1, using the formal procedure review process.

- PPM 3.1.3, "Plant Startup," and PPM 4.12.4.7, "Unintentional Entry Into Region of Potential Core Power Instabilities," incorporated the guidance contained in the NRCBs, and the BWROG letters through November 4, 1988 regarding recognition of core power instabilities, and mitigative actions if operating in a region of potential core instability.

- Lesson Guide 82-SQT-9202-L2, "Reactivity Mismanagement Events," incorporated the guidance contained in all the related NRCBs and BWROG letters through the March 18, 1992 BWROG letter. In addition, the lesson plan used several operating events to provide examples of what could occur that could lead to reactivity mismanagement, and the appropriate actions to take in each case.

However, the AIT found that the licensee had not fully implemented some other important generic guidance. The licensee's procedure for review of generic information was PPM 1.10.4, "External Operational Experience Review." This procedure required a documented review by all potentially affected departments to determine whether or not procedural changes or other action was appropriate. Information from the BWROG was not required to be reviewed under this procedure.

The information was, however, reviewed using a routine distibution of information list. This distribution list was used to provide information for documents not specifically covered by PPM 1.10.4. In the case of the March 18, 1992 BWROG letter, the licensee's representative on the Owners' Group sent the letter to personnel on the distribution list for BWROG information related to stability. This list included: Manager, Engineer Services; Manager, Nuclear Engineering; Principal Engineer, Reactor Engineering; former Supervisor, Shift Engineering; and STA Trainer.

These personnel then decided if additional action was required on the information provided through the distribution system. There was not any documentation of the decisions which they made to distribute this information. A licensee representative stated that none of the personnel on distribution considered that any prompt actions on the information were necessary. The STA Trainer did, however, incorporate the information into a lesson plan on power instability experience which he was updating. As a result of the failure to distribute this information throughout the organization, several errors occurred. These deficiencies were:

- The licensee's operating procedures did not contain the guidance from the March 18, 1992 BWROG letter discussed in Section 4.1 of this report.

- The SNE/STA stated that he conducted rod time testing and as many rod pulls as possible before shifting recirculation pumps to high speed, staying within a few percent of the 80 % rod line during these evolutions, contrary to the BWROG guidance. He further stated that this was normal practice, and no recent information had changed the practice.

- The Operations Procedure Supervisor was not aware of the March 18, 1992 BWROG letter, or its associated guidance, before the event. The Operations Procedure Supervisor is directly responsible to the

Operations Supervisor for evaluation of operating experience for incorporation into station procedures.

- The Plant Manager was not aware of the March 18, 1992 BWROG letter, or its associated guidance, before the event.

- Licensee personnel did not incorporate consideration of the filters on the APRMs and stability monitor, as directed in the BWROG letter dated November 3, 1988, and in NRCB 88-07 Supplement 1, into their consideration of APRM response to core power instabilities, or of the impact on the stability monitor. Licensee personnel also did not consider the effect of the filters in their event analysis until the effect of the filters was identified by the AIT.

- Plant Operating and Nuclear Engineering Procedures were not reviewed under the licensee's formal program for review of operating experience to ensure guidance and directions to operators were consistent with the latest generic information contained in the BWROG letter dated March 18, 1992. The AIT did note that the information from the BWROG letters of November 3 and 4, 1988, had been evaluated and incorporated in plant procedures under the licensee's previous program for formal review of external operating experience.

- The licensed operators and SNE/STAs were trained on the information contained in the BWROG March 18, 1992 letter, but the information was not provided to the Operations Department for evaluation for incorporation into procedures. The AIT found that the licensee's program for control of generic information had broken down: it permitted the procedures and training to conflict. Training was conducted on operating strategies that had not been coordinated with the Operations Department. The strategies were not the method by which the management expected the plant to be operated.[4]

5. Evaluation of Organizational Performance

5.1 Review of Operator and STA Performance

5.1.1 Assessment of Procedural Adequacy

PPM 3.1.3, "Reactor Startup from Hot Shutdown," provided guidance for conducting a plant startup from a hot shutdown condition. Once at power, PPM 9.3.12, "Power Plant Maneuvering," provided guidance to the SNE/STA and operators while changing power using control rods and recirculation pumps. In addition, PPM 2.2.1, "Reactor Recirculation System," provided detailed instructions for conducting a shift of recirculation pumps from slow to fast speed. PPM 4.12.4.7, "Unintentional Entry Into Region of Potential Core Power Instabilities" provided actions for the operator upon entry into areas of the power to flow map identified as core instability regions, or the onset of actual core power oscillations.

PPM 9.3.12, "Power Plant Maneuvering," contained guidance for operating in specified regions of potential core instability on the power to flow map, including use of the stability monitor for operating in Region C. No specific guidance was provided, however, for operating in areas where instabilities were not expected.

Based on a review of these procedures, analysis of available data from the plant computers, interviews with personnel, and observation of a Plant Oversight Committee (POC) meeting, the AIT determined that the procedures did provid several cautions which, if followed, would have reduced the probability that this event would have occurred. The specific guidance was:

- PPM 9.3.12, "Power Plant Maneuvering," also provided general principles and objectives that included: 1) flatten radial peaking, 2) maintaining a strong bottom peak, but having too large a bottom peak would prevent opening flow control valves after pump shift to fast speed, 3) not being overly aggressive when pulling shaper rods, with a peaking factor of 3.4 as an optimum value, considering recirculation pump vibration and fuel preconditioning limits, and 4) minimizing the amount of time spent at slow speed pumps when performing a rod set following a power reduction, due to concerns for xenon burnup after power increase.

As discussed in Section 5.1.2, the operating crew did not use this guidance. In addition, the AIT noted deficiencies with the following procedures:

- PPM 9.3.12, "Power Plant Maneuvering," provided guidance that allowed deviation from the Control Rod Withdrawal Sequence once above approximately 20 % power. This was provided based on margins between operating fuel rod powers and fuel preconditioning considerations. The procedure stated that deviations were considered necessary to account for power distribution and stability constraints, and to account for various core reactivity conditions and xenon inventories. Interviews with plant personnel demonstrated that deviation from the rod pattern above the low power setpoint (approximately 20 % power), was a common practice at WNP-2, with rod pulls based on calculations by the STA/SNE. The AIT was concerned because this practice resulted in large variations in stability during several startups, due to excessively peaked power distributions caused by the rod patterns selected by the STA/SNE.

- PPM 9.3.12, "Power Plant Maneuvering," provided guidance to complete most rod motions prior to shifting recirculation pumps to fast speed. Additional guidance was provided in this procedure to conduct rod withdrawal to increase power at the left side of the power to flow map. This required operating at close to the 80 % rod line for several hours while rod pulls were conducted, which was contrary to the guidance provided by the March 18, 1992 BWROG letter to avoid extended operation close to this area.

- Licensee procedure, PPM 2.1.8, "ANNA System Operation," required a scram if ANNA indicates that APRM oscillations peak to peak were greater than 10 %, and if APRM power range recorders indicated oscillations > 10 %. BWROG advisory letter 92030 dated March 18, 1992 stated that "it may be possible to violate the MCPR safety limit during a regional oscillation while the APRM's are oscillating less than 10 %. A regional oscillation ... may not always result in LPRM high or downscale alarms." When discussing training, the BWROG letter stated "this training should emphasize that a scram is required even if the magnitude is below 10 % on the APRMs and LPRM upscale and downscale alarms have not occurred." The letter also observed that oscillations may be occurring if APRM noise levels are two to three times normal peak to peak. Licensee personnel stated that, in the past, peak to peak noise has been about 2 % when the plant was at approximately 35 % power.

  The licensee's procedures did not incorporate this guidance .

- PPM 9.3.9, "Control Rod Withdrawal Sequence Development and Control" required calculated power distributions to be attached to each control rod order sheet. Paragraph 7.4.2 stated, in part:

  > Actual operating conditions, exposure histories relative to the target or preconditioning strategies may dictate using a different pattern than provided. With each rod sequence sheet, include as an attachment the target rod pattern, calculated power distributions and the target power distribution for the sequence review.

  However, the STA/SNE did not determine the projected power distributions for each rod sequence sheet as required by this procedure. Licensee representatives stated that the procedure was incorrect, and that they had not intended to perform these power distribution calculations.

- Although PPM 9.3.12 and PPM 9.3.9 provided qualitative guidance for controlling neutron flux peaking, the procedures did not define adjectives used to prescribe guidance. Examples of these guidance adjectives that did not have quantitative limits or examples included terms such as excessive, too large, optimal, overly aggressive and fierce. The lack of quantitative, and specific qualitative requirements on reactor power distribution parameters permitted the STA/SNE to implement an unstable control rod withdrawal pattern that generated the very high peaking factors described in Table 1. Although Technical Specification thermal limits provided quantitative power limits, these limits were based on full power operation, and did not effectively limit axial and radial peaking factors sufficiently to ensure stability during startup.

- Control rod sequencing procedures at other BWR sites were reviewed to determine if the practices at WNP-2 were typical. The AIT concluded, based on this review, that the practices at WNP-2 were typical.[3] This may indicate that some other BWR licensees have a similar vulnerability to instability when very skewed power distributions are used.

5.1.2 Evaluation of Operator and STA/SNE Performance

5.1.2.1 STA/SNE Performance

During the reactor startup, the STA/SNE monitored the reactor power profile and control rod withdrawal sequence to ensure that fuel limits were not exceeded, and that the flux profile would allow power ascension with minimal rod motion after pump shifting. He authorized several changes to the control rod withdrawal sequence specified by the approved Control Rod Withdrawal Order Sheets between 23:55:29 and 00:53:44. These changes preferentially withdrew control rods in the center of the reactor, while not withdrawing rods further out in the core. This caused an increase in radial peaking, which is represented in Table 1 by the increase in CMFLCPR between points 1 and 2.

The Core Maximum Peaking Factor (CMPF) and Core Maximum Fraction of Limiting Critical Power Ratio (CMFLCPR) parameters in Table 1 provide a measure of axial and radial neutron flux peaking. (These terms are defined in Appendix L.)

TABLE 1: Core Physics Parameters vs Time

POINT	TIME (HOURS)	POWER (MWth)	CMPF	CMFLCPR
1	23.55.29	1137	3.221	0.711
2	00:53:44	1231	3.274	0.924
3	01:05:44	1221	3.307	0.939
4	01:37:30	1212	3.500	0.933
5	01:53:29	1211	3.560	0.933
6	02:33:14	1153	3.696	0.884
7	02:57:14	1207	3.782	0.959
8*	03:00:14	1119	3.859	0.922

*Recirculation loop "A" FCV closed prior to this data.

The control rod pattern during this event was essentially held constant from 0105 until the event occurred; however, neutron flux peaking factors continued to change. Data at points 3, 4, 5, 7, and 8 of Table 1 were all taken from the same control rod pattern. The increase in CMPF over this time period was caused by xenon burnout resulting from the power increase that occurred during the previous seven hours. Xenon burnout acted to increase neutron flux peaking because it was burnt out of the core areas having higher power faster than areas having low power. The crew did not consider the increasing peaking factors to be a concern, and consequently, took no actions to limit them.

CMFLCPR is well correlated with the radial peaking factor; as it increased prior to the event, the radial peaking factor also increased. Points 3 to 7 of Table 1 show how CMPF and CMFLCPR increased prior to the event. The increases in CMPF show that axial peaking increased steadily between 0105 and 0258, the time of the event. The data in Table 1 was taken from computer calculations (MON runs) that were used by the crew to monitor control rod positions and reactor power distribution during the power increase prior to the event.

The AIT evaluated the guidance that was provided to the STA/SNE to control the reactor power distribution during startup (see also Section 5.1.1). The "Power Distribution Constraints" section of procedure PPM 9.3.12, stated that:

> Establishing too large of a bottom peak will prevent opening flow control valves (FCVs) to 20,000 and 24,000 GPM respectively following the shift to 60 Hz due to preconditioning or linear heat generation rate (LHGR) limits. Therefore, do not be overly aggressive when pulling shaper rods, especially on the first of two ramps. A (Core Maximum Peaking Factor) of approximately 3.4 usually results in the optimum rod pattern to meet both constraints during a xenon free rod set.

Even though the basis of this guidance was not core stability, controlling CMPF at a value of 3.4 would have reduced the core's neutron flux peaking factors, and improved stability. During the control rod positioning that was performed from 23:55:29 to 02:57:14, the crew did not attempt to limit CMPF to 3.4. The STA/SNE stated that he did not consider the CMPF values (from 3.22 to 3.78) during this time period to be excessive; he considered a higher CMPF to be better because it enhanced fuel conditioning and xenon buildup in the bottom of the core. The STA/SNE's strategy was endorsed by PPM 9.3.12, which stated that a strong bottom peak power distribution was desirable during startup.

However, the guidance in PPM 9.3.12 to maintain a strong bottom peaked power distribution during startup was tempered by other guidance within the procedure that warned of problems caused by excessive peaking. Section 7.1.5.e, "Xenon Constraints," specifically identified that prolonged operation with slow speed (15 hertz) recirculation pumps would lead to high peaking factors induced by xenon burnout. This paragraph stated, in part:

> The more time spent at low power, the more fierce the xenon burn following power increase. Normally, a rod set can be performed in approximately 2 hours. .... If more time than this is spent at low power, the resulting xenon burn may make it difficult to meet peaking factor and RRC (Reactor Recirculation) pump operation constraints.

Emphasis was also provided by section 7.1.3.c, "Minimize Excessive Neutron Flux Peaking." This section called attention to the Technical Specification requirement to adjust APRM trip setpoints when the total peaking factor exceeded rated values. This paragraph stated, in part:

> If the core total peaking factor increases above its design value, then per Technical Specifications (3/4.2.2), the APRMs must be adjusted to make the trip setpoint more conservative. .... This may result in an inability to attain the rated load line before a rod block is received. Consequently, care should be taken to prevent the occurrence of excessive peaking.

In summary, the AIT concluded that the STA/SNE gave recommendations to the licensed operators which caused the peaking factors prior to this event to be excessive from a stability perspective.

5.1.2.2 <u>Licensed Operator Performance</u>

The AIT interviewed the five licensed operators, and the STA/SNE, that were on shift at the time of the event. These interviews provided information that was used to confirm data captured by the Transient Data Acquisition System (TDAS), and also identified the following points:

- The Shift Nuclear Engineers (SNE) were highly respected by the operators, who felt that their qualifications were outstanding.

- The operators felt that controlling the reactor's power distribution profile was solely the SNE's responsibility. The operators ensured that Technical Specifications were not violated, but generally viewed the SNE's authority in determining flux shape as final.

- None of the operators could recall the classroom training they had received in June, 1992, on the Boiling Water

Reactor Owners' Group (BWROG) letter, "Implementation Guidance for Stability Interim Corrective Actions," dated March 18, 1992 (see Section 5.1.3).

- The operators considered the area of core instability to be well defined. That is, they felt there was no possibility of the reactor becoming unstable so long as it was operated outside this area (see Section 5.1.3).

- The operators consistently described the post-scram events as an easy, smooth, and uncomplicated reactor shut down.

The inspectors found that there were no administrative controls requiring peer review of SNE changes to prescribed control rod order sheets. Although the CRS had been formally designated as the Reactivity Manager, his duties were broadly defined. PPM 1.3.1, "Conduct of Operations," paragraph 5.4.20 describes these responsibilities as:

> Responsibilities include ensuring a conservative approach to operations involving core reactivity changes.

The AIT concluded that the rapport between the STA/SNE and the operators, the lack of procedural requirements, and particularly the lack of specific requirements regarding startup flux shaping, explained the absence of operator review of changes by the STA/SNE to the prescribed control rod withdrawal sequence.

The AIT concluded from a review of the sequence of events and interviews of the operators that the actions taken by the control room crew to terminate this event were prompt, effective, and complied with the licensee's procedures. Prior to the event, the control rod sequence used by the crew caused very high neutron flux peaking, and did not conform to guidance provided by the licensee's procedures. Procedural guidance and requirements regarding power distribution during reactor startup were non-specific, and relied upon the SNE to interpret broad, qualitative guidance.

5.1.3 Operator and STA Training Effectiveness

The AIT evaluated the effectiveness of operator training related to information, procedures, and operations affecting the event. The AIT reviewed operating procedures, training attendance records, lesson plans, examinations, interviews, and observation of a video taped session of a classroom lecture. The operating procedures reviewed are listed in Section 5.1.1, 5.1.2 and Appendix K of this report. The lesson plan reviewed was Lesson Code 82-SQT-9202-L2, "Reactivity Mismanagement Events," dated May 27, 1992, in the STA Continuing Training program.

Based on this information, the AIT found:

- The training department used the information available in the generic correspondence described in section 4.1 of this report to develop lesson plans and train operators on relevant operating experience. The lesson plan approved on May 27, 1992 incorporated information from the March 18, 1992 BWROG letter.

- The training session from a June 2, 1992 requalification class which the AIT observed on video tape provided very good coverage of the information from the March 18, 1992 BWROG letter.

- All licensed operators on shift at the time of the event had attended requalification training that included this training on instability. All operators took and passed written examinations that included the instability topic. However, the AIT found that none of these licensed operators could recall having attended training that addressed the March 18, 1992 BWROG letter concerns regarding core instabilities, or that there was new guidance from the BWROG. None of the operators recalled any of the specific guidance from the BWROG March 18, 1992 letter.

- The training conducted was not based on changes to procedures. The guidance was not sent to anyone in the Operations Department, and no procedure changes were evaluated or initiated based on the March 18, 1992 BWROG letter. The guidance from the March 18, 1992 BWROG letter was incorporated into SNE/STA and licensed operator training because the STA trainer decided to include it in the reactivity mismanagement lecture. While the initiative demonstrated by the STA trainer was commendable, the AIT found that the lack of a formal review process for this important guidance was a serious weakness.

- The AIT was also concerned with the potential for confusion by the shift staff, since training covered operating concerns that were not incorporated into plant operating procedures. The AIT determined that confusion did not occur, since the shift staff did not recall the information provided in the training lectures covering the guidance in the March 18, 1992 BWROG letter.

- The SNE/STA on shift at the time of the event had attended prior requalification training on instability events that did not include information from the March 18, 1992 BWROG letter.

- The licensee used a systems approach to training to develop topics and lesson plans. This approach was procedures-based, meaning that task lists were developed from procedures, and job-task analyses were performed to determine the knowledge, skills, and abilities necessary to complete the tasks. Lesson plans were then developed to incorporate the job-tasks into an appropriate training setting. This process was not followed for the May 27, 1992 lesson plan discussed above.

The AIT concluded that the training conducted on potential core instabilities was ineffective: the SNE/STA on shift did not attend training on the latest information available from the BWROG, and none of the personnel on shift at the time of the event who were trained on the BWROG guidance could recall either having received the training, or the details contained in the lesson plan.

5.2 Assessment of Engineering Performance

5.2.1 Evaluation of Core Design

The WNP-2 Cycle 8 core loading consisted of 76 fresh SNP 9x9-9X fuel assemblies, 112 once burned SNP 9x9-9X assemblies, 8 once burned SNP 8x8 assemblies, 4 twice burned ABB Atom (SVEA-96) lead fuel assemblies, 4 twice burned GE-11 (9x9) lead fuel assemblies, 144 twice burned SNP 8x8 assemblies, 4 three times burned SNP 9x9 lead fuel assemblies, 132 three times burned SNP 8x8 assemblies, 152 four times burned SNP 8x8 assemblies, 120 five times burned SNP 8x8 assemblies, and 8 six times burned SNP 8x8 assemblies. This multiplicity of fuel types and power histories was somewhat unusual. Licensee personnel stated that it related to the Supply System's preference for annual refueling outages (with small fresh fuel batches) and a continuing evaluation of different fuel designs and fuel suppliers for optimum fuel economy.

The Cycle 8 core loading and fuel design were selected by the Supply System to maximize end of cycle (EOC) core reactivity. The projected control rod patterns and licensing evaluations of shutdown margin, hot excess reactivity, and thermal margins were performed by Siemens Nuclear Fuel (SNP) Corporation, and are presented in the Cycle 8 Fuel Cycle Design Report. Results of the SNP evaluation of system transient events are presented in the "WNP-2 CYCLE 8 PLANT TRANSIENT ANALYSIS REPORT," EMF-92-039. The analyses were performed using analytical methods which have been reviewed and approved for generic applications by the NRC staff. The Cycle 8 safety analyses were performed under provisions of 10 CFR 50.59 and were not submitted for review by the NRC staff. The AIT performed an audit of the Cycle 8 safety evaluation and identified the following issues discussed below:

5.2.1.1 Core Stability

The Cycle 8 core stability evaluation by SNP consisted of three decay ratio calculations using their single channel time domain code COTRAN.

TABLE 2: DECAY RATIO CALCULATIONS PERFORMED BY SNP

Power/% Flow	Operating Map Location	Decay Ratio Found/ (SNP Design Acceptance Criteria) Cycle 8	Cycle 7
65/45	Intercept of 45 % flow exclusion boundary and APRM rod block line	0.41/(<.75)	0.42
47/27.6	Two pump minimum flow intercept with 100 % rod pattern line-Region A exclusion boundary	0.77/(<.90)	0.86
42/23.8	Natural circulation flow intercept with 100 % rod pattern line-Region A exclusion boundary	0.64/(<.90)	0.81

The .75 DR acceptance criteria was based on a May 10, 1984 NRC safety evaluation of the licensing topical report XN-NF-691P, "Stability Evaluation of Boiling Water Reactor Cores," and its Supplement 1. The .9 DR acceptance criteria was an SNP guideline.

The 1984 review also concluded, separately, that COTRAN calculated core decay ratio values greater than 0.75 indicated potential core instability because of code calculational uncertainties. The staff concluded that the methodology was acceptable for licensing of reload fuel with the condition that acceptable technical specifications were required to restrict operation if the calculated decay ratios exceeded 0.75.

The nature of the technical specification limitations on permissible operation was later modified to be consistent with Generic Letter (GL) 86-02. GL 86-02 directed licensees to evaluate each core reload to assure that it was typical of previously evaluated cores which have acceptable stability margin. Following the 1988 LaSalle instability, NRC Bulletin 88-07 and its Supplement 1 provided further guidance to reduce reliance on decay ratio calculations for avoidance of power oscillations, to improve training and procedures, and to verify the adequacy of instrumentation relied on for procedural response to oscillations. Supplement 1 directed that for proposed new fuel designs, the stability exclusion region boundaries (based on GE fuel designs) should be reevaluated and justified based on any applicable operating experience.

On March 18, 1992, the BWROG transmitted a letter to all owners with recommendations for improved implementation of Supplement 1 based on the insights obtained from the work of the BWROG Stability Committee. The letter stressed the need for greater caution when operating near the exclusion boundaries. Reexamination of procedures and training to reflect uncertainties in the definition of power and flow stability exclusion boundaries was suggested.

During the licensing review of Advanced Nuclear Fuel (ANF) 9x9 fuel, both ANF (the predecessor of SNP) and the NRC concluded that the design was less stable than existing designs of 8x8 fuel in operating reactors. During its introduction in the Susquehanna BWRs, the staff required initial startup stability monitoring of each core reload until the transition to a full 9x9 core was complete. On the introduction of the SNP 9x9-9X design in Cycle 7 of WNP-2, stability decay ratios of 0.42, 0.86, and 0.81 were calculated using COTRAN for the respective power/flow statepoints identified in the preceding tabulation. These were not provided to the NRC at the time. After the August 15 event, LAPUR analyses by the staff and hydraulic stability evaluation by SNP (based on the ratio of two phase versus single phase pressure drop) indicated that the 9x9-9X fuel was less stable than the ANF 9x9, and less stable than other fuels in US BWRs.

The reduced (>.75 decay ratio) stability margin which was calculated by SNP for Cycle 7 and 8 resulted in operation with reduced thermal margin during power oscillations (in fact, the CPR operating limit was exceeded during the oscillations of August 15). The licensing bases assumed that a design basis accident would not be initiated from a condition which exceeded the CPR operating limit. The licensee and SNP accepted the reduced stability margin for Cycle 7 and 8 without reviewing the possibility of initiating a design basis accident with this reduced stability margin and potentially, reduced CPR.

The AIT concluded that the application of 9x9-9X fuel in WNP-2 should have received more specific licensing attention to the reduced stability margin, and potentially, may have been an unreviewed safety question.

5.2.1.2 <u>Core Nuclear and Thermal-Hydraulic Design</u>

Historically, a major design goal for new fuel types in an existing core has been to match the hydraulic resistance of the existing fuel in order to maintain thermal-hydraulic compatibility of the mixed core during power operation. Differences in hydraulic resistance tend to starve flow from the more resistant fuel and increase the uncertainty in the calculated core inlet flow distribution. While there is no regulatory criterion governing the acceptability of the match, the approximately 5 % difference in pressure drops (measured from the inlet orifice to the top of the core support plate) between the WNP-2 original GE P8x8R and the initial reload SNP 8x8-2 is typical. The pressure drop mismatch between the SNP 8x8 and the SNP 9x9-9X is about 10 %, which appears to be undesirably large. The Cycle 8 core design used the relatively unstable fresh SNP 9x9-9X fuel in a mixed core loading pattern which maximized the flow mismatch with neighboring once and twice burned SNP 8x8 fuel in regulating rod control cells. The core loading with fuel of several different types and exposure histories also resulted in a core unrodded power distribution with radial peaking which is greater during the early core life than it is in other BWRs fueled by SNP. However, unrodded radial peaking was also compared to LaSalle Unit 2, Cycle 4, and found to be comparable. WNP-2 was higher for the first three GWd/MTU and then slightly lower than LaSalle.

This core nuclear design may make the core more vulnerable to high peaking factors during core power maneuvering by injudicious selection of control rod patterns. As discussed in Section 3.3 of this report, extremely skewed radial and axial peaking factors which occurred during startup maneuvers were the direct cause of the WNP-2 instability. In addition, the mixed core design, and the SNP 9x9-9X fuel, in particular, may have made the core more vulnerable to both the core-wide and out-of-phase modes of instability (see Section 3.2).

5.2.1.3 <u>Exclusion Region Boundaries Assumptions</u>

The AIT considered whether appropriate design attention had been given to maintaining the core power distributions assumed consistent with the technical specification exclusion boundaries for WNP-2. The BWROG methodology for exclusion boundary calculations is described in BWROG NEDO 31960, "BWROG Long-Term Stability Solutions Licensing Methodology," Supplement 1, dated March 1992. That methodology assumed a bottom peaked axial peaking factor of 2.0 in Node 3 of 24, and a radial peaking factor based on an End of Cycle (EOC) Haling calculated power shape (typical radial peaking factor of 1.5 for BWR 5s). However, the exclusion boundary used for the interim stability solution required by NRCB 88-07 and its Supplement 1 was based on operating experience and was considered to be conservative. Therefore, no limits were specified on power distribution. Nevertheless, the AIT concluded that sufficient information was available to indicate a relatively unstable core, and neither the licensee or SNP gave appropriate attention to the design and operation of the core to assure its stability.

5.2.1.4 Reduced Core Flow Capability

The licensee reported that maximum recirculation flow capability of WNP-2 had been reduced from 106 % to 96 % of nominal because of heavy crud deposits in the jet pumps. The pumps were inspected by GE during the recent refueling outage and there was no evidence of cracks or other structural defects. The source of the crud was believed to be metallic corrosion products from the condenser tubes, though the evaluation was incomplete at the time of the inspection. The AIT viewed photographs of the jet pumps showing heavy deposits at the venturi entrance with gradual reduction in crud thickness proceeding down the pump.

The AIT also noted that the natural circulation flow line for WNP-2 is significantly lower than other flow control valve (FCV) reactors (for example, about 5% less than the LaSalle units). This may be due to differing hydraulic characteristics, and makes WNP-2 more vulnerable to low flow instability.

The AIT noted that the plant transient analysis was performed at the 104 % power-106 % flow point which SNP confirmed to be conservatively applicable to rated power and flow conditions, but did not address the 96 % degraded flow capability that was known to exist. However, the licensee claimed to have considered the applicability of the existing analyses to the reduced flow condition. Though not documented, this review concluded that the existing analysis was bounding for permissible power/flow map operation based on the techniques employed for the operating MCPR limit determination. Later, SNP analysis confirmed that the 96 % flow condition was bounded by the existing analysis.

The AIT concluded that the WNP-2 Cycle 8 core design contributed to the instability of the core. Also, the impact of the SNP 9x9-9X fuel (initially loaded for Cycle 7) on the stability of the WNP-2 core was not adequately reviewed by either the designer or the licensee. Nevertheless, the AIT and the licensee concluded subsequent to the August 15, 1992 event, that the core could be operated safely by conforming to the compensating operating restrictions and precautions specified in the licensee's August 29, 1992 letter to the NRC for future low flow startup and shutdown operations. These precautions included continuous monitoring with the ANNA stability monitor below 50 % flow with power >25 %. [5, 6]

5.2.2 Evaluation of Core Design Quality Assurance

The AIT reviewed the licensee's and the fuel vendor's core design quality assurance, and staffing as related to the August 15, 1992 event.

5.2.2.1 Review of Supply System Core Design Process

Approximately six months prior to a refueling outage, the licensee begins planning for the next cycle reload. A fuel vendor is determined by the licensee and contracts are prepared. For the Cycle 8 reload, the licensee selected SNP as its fuel vendor and reload designer. A Final Energy Notice was issued by the licensee on August 28, 1991 that established the final energy requirements and the design criteria for the reload. The design criteria established the licensing, operational and economic requirements for the reload. The Energy Notice established the thermal limits, reactivity limits and the design margins for those limits. The notice was revised on November 22, 1991 to update the cycle energy requirements based on Cycle 7 fuel utilization.

Licensee personnel stated that during the cycle reload design process, Nuclear Engineering personnel were in frequent communications with the fuel vendor. Because of the close proximity of the fuel vendor, the licensee often had direct participation in the design process despite the design responsibility having been contracted to the fuel vendor. Upon completion of the design, the fuel vendor provided the licensee with several design analyses.[7]

Upon completion of the design and receipt of the various design analysis reports, the licensee performed a sampling design review for each reload. For cycle 8 reload, a sampling design review was performed on the fuel mechanical design, thermal-hydraulic design, safety and transient analysis and core monitoring. A design review plan was prepared and approved; a design review checklist was prepared and approved; the results of the design review were documented and questions and concerns resolved. The licensee's sampling design review, in essence a technical audit, appeared to be penetrating in the areas reviewed. However, the Cycle 8 design review, and other previous cycle design reviews performed by the licensee, did not appear to have included a review for core stability.

A Core Operating Limits Report (COLR) was prepared by the licensee upon review and evaluation of the fuel vendor Cycle 8 design reports noted above. The COLR described the Cycle 8 reload, provided the design thermal limits, and ascertained that applicable limits of the plant safety analysis were met. The licensee performed a 10 CFR 50.59 review for the COLR and submitted a change request to the WNP-2 Technical Specifications (TS) to reflect the Cycle 8 reload. Amendment 109 to the facility operating license provided the NRC evaluation and approval of the proposed TS changes.

The AIT found that the licensee had performed appropriate sampling design reviews, appropriate evaluation of fuel design analysis results and report, and had appropriately utilized the design information for the reload in establishing the cycle thermal limits. However, the AIT also found that the licensee's reviews had not thoroughly reviewed Cycle 8 core stability.

5.2.2.2 Evaluation of Core Design QA Effectiveness

The licensee selected SNP as both the Cycle 8 reload designer and as the fuel vendor. SNP was an approved vendor on the licensee's approved suppliers list based on both Supply System audits and industry audits. SNP had an NRC approved Quality Assurance (QA) program. Revision 25 to its QA Topical Report was reviewed by the NRC and approved by the NRC in a letter to SNP dated February 18, 1992.

The AIT performed interviews, sampling procedure reviews, and sampling analysis reviews to determine the adequacy of the fuel vendor's design process for the WNP-2 Cycle 8 reload as it relates to the August 15, 1992 event. Several weaknesses were observed that appeared to have been missed opportunities to minimize the occurrence of the core instability that occurred. The weaknesses observed were as follows:

- Discussions with fuel vendor design management personnel indicated that they were not aware of the BWROG March 18, 1992 implementation guidance and stability interim corrective actions.

- A Cycle 8 stability analysis (Calculation E5072-N06-3 dated January 31, 1992) was performed by the fuel vendor as part of analyses for the cycle 8 reload report. Four points, approximately in the periphery of the stability exclusion zone, were analyzed to demonstrate core stability. However, the analysis was based on end of cycle conditions and with all control rods withdrawn. No consideration of rod patterns, and the consequent effect on stability, was required. No analysis of the effects of various rod patterns was performed. No recommendations were provided for rod patterns that may be required during operation in the proximity of the stability exclusion zone.

- An evaluation was performed by the fuel vendor for "final feedwater temperature reduction" with thermal coastdown toward the end of Cycle 8. However, the evaluation did not include a specific stability analysis for the Cycle 8 reload. A stability analysis performed for Cycle 3 was used as a basis for the stability evaluation despite the differences in flow characteristics of the two cores.

The AIT's interviews and procedure and analysis sampling review also indicated additional weaknesses that did not relate directly to core stability but which were design process discrepancies:

- The WNP-2 cycle 8 reload design group consisted of seven engineers and one technician. The design group performed independent verification of the various analyses required for the reload design. However, the independent verifier and the analysis originator frequently interchanged functions for various analyses due to the limited number of people in the group. The effectiveness of the independent verification function was further minimized by independent verification of an individual's work on an analysis that utilized inputs from a different analysis that had been performed by the same independent verifier. For example, this was the case for the cycle 8 reload stability analysis calculation.

- The fuel vendor had programmatic procedures for its design process that appeared to meet the requirements of 10 CFR 50 Appendix B and ANSI N. 45.2.11, "Quality Assurance Requirements for the Design of Nuclear Power Plants." However, the vendor did not have specific detailed procedures for performance of various specific analysis for the core reload design. In addition, vendor programmatic procedures allowed design calculations to remain with the designers, uncontrolled, approximately six months after the cycle startup. At the time of the inspection, the Cycle 8 stability calculations were still on the originator's desk.

5.3 Evaluation of Equipment Performance

The AIT reviewed the operation of the equipment listed below to determine any contribution of the equipment to the event. Equipment that potentially had a significant effect on core stability and monitoring was chosen for review. The AIT reviewed the Shift Manager's log, Control Room Operator's log, TDAS data, alarm print-outs, control room deficiency tags, associated maintenance work requests, associated problem evaluation reports, and the equipment history computer data base. The AIT interviewed licensed operators, STAs, design and system engineers, and other members of the licensee's engineering and plant staff to determine equipment performance prior to, and during the event, and to determine their potential contribution to the event. The AIT concluded that no equipment discrepancy was a direct contributor to the event.

5.3.1 Recirculation Flow Control System

Flow Control Valves (FCV) - 2-RRC-FCV-60A and 2-RRC-FCV-60B and Recirculation Pumps 2-RRC-P-1A and 2-RRC-P-2B

The reactor was at approximately 36.4 % power and core flow was at approximately 30.5 % flow, both FCVs were in their full open position and both recirculation pumps were operating at 15 HZ (slow speed) prior to the event. In preparation for shifting the recirculation pumps to 60 HZ (fast speed), FCV-60A was closed to its minimum open position in accordance with operating procedures. TDAS data indicated appropriate recirculation flow response to the valve manipulation from approximately 9600 gpm to approximately 1000 gpm. Core flow decreased accordingly. Noticeable indication of power oscillations shown in the TDAS plots started when recirculation train A flow decreased to approximately 2000 gpm and FCV-60A was approximately 35 % open. FCV-60B was left in the full open position through the event and TDAS data indicated a relatively constant and appropriate train B recirculation flow of approximately 10,000 gpm. Both recirculation pumps were at their 15 HZ slow speed prior to and throughout the event.

5.3.2 Jet Pumps

TDAS data indicated that flow through the A Train jet pumps was approximately 17.37 million pounds/hr and through the Train B train jet pumps was approximately 16.12 million pounds/hr prior to closing FCV-60A. Flow through the Train A jet pumps decreased to approximately 10.4 million pounds/hr after the onset of power oscillations. Prior to the event, the licensee had been aware of jet pump fouling at WNP-2, had instrumented the jet pumps, was monitoring the pump performance, and was considering possible corrective actions. The licensee considered that jet pump fouling was not unique to WNP-2 and was aware of other BWRs with similar conditions. The net effect of the jet pump fouling at WNP-2 has been a decrease of core flow such that WNP-2 is currently only capable of approximately 96 % core flow. In addition to TDAS data, the licensee reviewed daily jet pump surveillance data since July 18, 1992. The licensee review revealed no significant jet pump performance variance that directly caused the event. The TDAS jet pump flow data confirmed the licensee's evaluation. (See also the discussion in Section 5.2.1.4)

5.3.3 Feedwater Control System

Licensee personnel stated that the feedwater heaters' water level control system was in automatic prior to and during the event. They also stated that no abnormalities of the feedwater heaters, feedwater temperature and feedwater flow were noted prior to or during the event. A review of TDAS data confirmed the interview information. Feedwater loop A temperature was approximately 309° F and loop B was approximately 311° F prior to the event. Feedwater flow was approximately 2.2 million pounds/hr prior to the event. Turbine bypass valve testing and control rod drive valve timing were performed a few hours prior to the event. Feedwater flow changes occurred as expected but resulted in no significant flow and temperature variance that contributed to the event.

5.3.4 Turbine Control System

Supply System operations and technical staff indicated that no abnormalities with the Digital Electrohydraulic Control (DEH) system that had an effect on the event were noted prior to the event. The DEH system controls reactor pressure. TDAS data confirmed that reactor pressure remained relatively normal at approximately 956.6 psig prior to the event.

5.3.5 Loose Parts Detection System

Prior to and during the event the loose parts detection system was inoperable as indicated in the Shift Manager's Log. Technical Specification 3.3.7.10 requires that the loose parts detection system be operable during startup and operation. The specification also states that TS 3.0.3 and 3.0.4 were not applicable to this situation. The action statement for TS 3.3.7.10 requires that with one or more loose parts detection channels inoperable for more than 30 days, a special report be submitted to the commission. The AIT found that, in this case, the licensee was in compliance with its Technical Specifications since the 30 day limit had not been reached. No data relevant to the event was available from the loose parts detectors since the system was inoperable.

5.3.6 Stability Monitor

TS 3.2.7 requires the stability monitoring system to be operable when operating in the region of potential instability defined in the technical specifications for various core flow and core thermal power conditions. Supply System personnel indicated that WNP-2 had never operated in the defined region of instability and consequently had never had to have the stability monitoring system operable. The monitor had been operated previously, however, for testing and, on one occasion, for informational purposes. The STA on-shift at the time of the event stated that WNP-2 was not in the Technical Specifications defined region of instability prior to and during the event. Because of this condition, the stability monitoring system was not operable, nor was it turned on, prior to and during the event.

The stability monitoring system at WNP-2 was purchased from SNP and installed in WNP-2 as a non-safety related system. The SNP stability monitor, called the Advanced Neutron Noise Analysis (ANNA) monitor, was evaluated by SNP for suitability for use at WNP-2 based on the LPRM and APRM inputs to ANNA.

Interviews held with SNP personnel indicated that SNP had recognized that the LPRM and APRM inputs to ANNA would have a 0.3 HZ filter. SNP considered the effects of the filtering of the LPRM and APRM inputs to the stability monitor. They tested for the filtering effect using a single sine wave, and determined that the filtering would have no effect on the decay ratio (DR) determined by ANNA. However, the amplitude of oscillations determined by ANNA would be affected. SNP determined that the amplitude warning and alarm setpoints for ANNA should be changed, and made the changes accordingly.

However, SNP personnel indicated that the change was only verbally communicated to Supply System personnel, but not documented in writing. The 10 percent peak to peak oscillation amplitude ANNA alarm was reset by SNP to 4.2 percent due to the filtering of the input data. That is, a 10 percent amplitude oscillation would be shown as a 4.2 percent amplitude oscillation by ANNA. WNP-2 surveillance procedure 7.4.2.7.3, Revision 2, dated December 9, 1991, "Core Stability Monitoring," paragraph 8.2 step 4, required a manual SCRAM if both ANNA APRM peak signal oscillations were greater than 10 percent, and one or more APRMs indicated a peak to peak amplitude greater than 10 percent of rated. Steps 7 and 8 of the procedure required various steps to be taken should the DR exceed 0.6. The procedure did not appear to have accounted for the 0.3 HZ filter nor the SNP evaluation and subsequent change to the ANNA alarm that was necessitated by the 0.3 HZ filter.

Licensee personnel stated that licensee post-installation testing of ANNA was performed by putting digital data in to ANNA, downstream of the 0.3 HZ filter. Consequently, the post installation testing did not identify the variance that the filter caused. The post installation testing of ANNA did not appear to have been comprehensive, in that it did not appropriately test the total system's interaction with the input data.

Subsequent to the event, the Supply System recovered TDAS data and reanalyzed the data to determine what indications may have been identified by ANNA. LPRM and APRM data and corresponding licensee generated graphs were reviewed by the AIT. The AIT noted differences in signals for two of the 18 LPRMs and a potential discrepancy in the data due to filtering. Subsequent licensee inspection of the LPRM and APRM inputs to ANNA determined that LPRM 32-09 and 32-17 had a 5 HZ filter installed in lieu of the required 0.3 HZ filter. Licensee and SNP evaluation of the processing of ANNA data based on independent processing both at WNP-2 and at SNP confirmed that the WNP-2 ANNA, except for the two LPRM feeds that had a 5 HZ filter, was working appropriately. Furthermore, event data processed through ANNA after the event indicated a higher than normal DR would have been shown by ANNA a few minutes prior to the event had ANNA been operating. Also, the SNP evaluation determined that the 0.3 HZ filter made a negligible difference when the DR was 1 but had an increasing effect as decay ratio decreased (up to a 0.27 DR difference with a 5 HZ filtered DR of 0.88). The .3 HZ filter had a significant effect on ANNA DR output at less than 1 DR and would not have been fully effective for TS required stability monitoring had WNP-2 been previously in the TS defined region of instability. As a result of the evaluation, SNP recommended, and the licensee accomplished, a change of the 0.3 HZ filters to 5 HZ.

The ANNA hardware and software were not safety grade. ANNA does not provide an audible alarm; for this reason, the licensee has modified their procedures to have a dedicated ANNA operator to monitor ANNA whenever operating above 25 % power and flow and below 50 % power and flow.

5.4 Evaluation of Licensee Event Investigation

The licensee initiated two efforts to investigate the August 15, 1992 event: a technical analysis and a root cause analysis. Technical analysis of the event has been completed. However, efforts to further define the core instability region and other longer term corrective actions were continuing.

At the conclusion of this inspection, the licensee's root cause analysis had not been completed. A preliminary executive summary of this analysis was reviewed by the AIT.

5.4.1 Technical Analysis

The licensee presented an initial technical assessment of the event to the AIT on August 17, 1992. The assessment was flawed in that it did not consider that all but two of the LPRM data recordings had been attenuated by about 75 % (see Section 5.3.6). The licensee used data from one of these LPRMs, located at core position 32-09A, to represent the hot channel (worst case) for analysis purposes. The licensee analyzed LPRM 32-09A data and determined that no Operational Minimum Critical Power Ratio (OMCPR) limits had been violated. The licensee did not attempt to explain why the peak oscillations appeared to take place on the core periphery, instead of at the center of the core where the highest neutron flux density was located. The licensee later corrected and reanalyzed the LPRM data for the correct hot channel at the center of the core. The licensee's reanalysis identified that the OMCPR limits were exceeded, but not the Safety Limit CPR. The attenuation of LPRM data appears to have been identified in parallel by the AIT and licensee staff. Other aspects of the licensee's initial analysis appeared to be acceptable.

The licensee continued the technical assessment of the event through the second week of the AIT's site visit with assistance from their fuel contractor, Siemans Nuclear Fuel (SNP). Short term corrective actions initiated by the licensee's technical investigation included:

- Core stability was analyzed for a variety of different power distributions near the critical power/flow point where recirculation pumps are shifted to fast speed.

- Stability comparisons were made between the August 15, 1992 event and previous startups.

- Operational limits were established for plant parameters that effect reactor stability in order to reduce the likelihood of instability.

- Attenuation effects of filters on the LPRMs were analyzed. The 0.3 hertz filters on the LPRM inputs to ANNA (Advanced Neutron Noise Analysis) were replaced with 5.0 hertz filters.

- Procedures were revised to include the cautions and limitations that resulted from these analyses.

The AIT concluded that the licensee's technical investigation adequately determined the immediate causes of this event.

5.4.2 Root Cause Analysis

The licensee's preliminary root cause analysis executive summary adequately identified what happened during this event; however, it did not develop an explanation why these events were not prevented by the licensee's staff. The AIT identified several areas that were not addressed in the executive summary:

- Inadequate implementation of BWROG guidance on instability.

- Ineffective operator training and STA/SNE qualification regarding instability.

- Failure of the crew to compensate APRM alarms/trips for high flux peaking factors (see Section 6 for further discussion).

- Failure to identify technical assumptions used in safety analyses and core stability analyses, and incorporate this limitations into operating procedures.

5.5 Assessment of the Licensee's Emergency Notification Process

PPM 13.1.1 "Event Classification," was the licensee's implementing procedure for emergency event classification. PPM 13.1.1 contained no specific direction on classification of a power oscillation event. However, PPM 13.1.1 defined a Notification of Unusual Event (NOUE) as "A condition at the plant or its surroundings, that threatens the normal level of plant safety, or an event where an increased awareness on the part of plant operating staff is warranted."

Based on this information, the Shift Manager declared a NOUE due to "core instability."[8]

The AIT determined that the licensee's event classification was proper for the August 15 power oscillation event, and that sufficient guidance was provided in the licensee's procedures for this determination. All notifications appeared to be prompt and informational. The AIT concluded that termination of the event when plant conditions were stable, with the operators executing a controlled cooldown of the plant, was conservative.

6. Description of Flow Biased ATWS Trip Setpoint Errors

During a review of the data provided by the licensee, the AIT noted that the thermal limits printout at 0153 on August 15, 1992 indicated that the core T-factor (a measure of flux asymmetry defined by TS 3.2.2) was calculated to be 0.793. However, in response to this, the licensee did not perform the flow biased setpoint adjustments or APRM gain adjustments as required by TS 3.2.2 whenever the T-factor was less than 1. In addition, five other thermal limits printouts, between 0053 and 0300 on August 15, 1992, also indicated that the adjusted flow biased setpoints should have been in effect. The lowest value calculated for the T factor was 0.746, just prior to the closure of the A loop recirculation flow control valve.

The AIT's review of licensee records indicated that PPM 7.4.2.1, "Power Distribution Limits," had not been performed on August 14 nor August 15. (PPM 7.4.2.1 was the licensee's implementing procedure for determining margin to the thermal limits, for adjustment of the flow biased setpoints discussed above.) The STA stated that TS 3.2.2 had always been interpreted by the Supply System staff to allow 12 hours after achieving 25% power for performance of PPM 7.4.2.1. The STA also stated that although the computer printouts indicated that the APRM flow biased setpoints required adjustment, this adjustment could also be delayed for 12 hours. The AIT concluded that this was an incorrect interpretation of the TS by the licensee.

PPM 3.1.2, "Reactor Plant Cold Startup," step 5.8.8 stated "Notify the STA to verify APLHGRs, LHGRs, MCPRs and APRM setpoints." The CRS stated that he signed the step off because he notified the STA. The Operations Manager stated that he agreed with the CRS, and that the responsibility for completing this step rested with the STA. The AIT concluded that this was a poorly written step that could have provided much more clear delineation of responsibility for completing a Technical Specification requirement. Step 5.8.8 was rewritten by the licensee prior to completion of the AIT inspection to require the STA's signature in PPM 3.1.2 that 7.4.2.1 has been satisfactorily completed prior to exceeding 25 % power.

The AIT calculated that the proper scram setpoint should have been 43.6 percent power, and the rod block/APRM Upscale alarm setpoint should have been 36.9 percent. The instrumentation for the flow biased scram inserts a six second time delay to simulate the thermal time constant of the fuel, and therefore did not provide scram protection for a power oscillation event. The flow biased rod blocks and APRM Upscale alarms do not have a time delay. These trips apparently would have been received earlier in the event had the setpoints been lowered, providing the operators with earlier information to diagnose the onset of power oscillations. In addition, as a result of these alarms, operators might not have increased reactor power to 36.4 percent, due to the close proximity to the rod block trip point.

APPENDIX A

Notes

(1) These numbers were estimated in a coarse manner from the following facts derived from Fig. 1:

- LPRM 32-17-C average value was 19 W/cm^2
- LPRM 32-17-C peak value at 02:59.00 was 21 W/cm^2
- LPRM 32-17-C peak value at 02:59.30 was 24 W/cm^2
- There were 15 oscillations in the 30 seconds between those measurements, indicating a 2 second oscillation period (0.5 Hz).
- Therefore, the decay ratio was estimated to have been

$$DR = \left(\frac{24-19}{21-19}\right)^{\frac{1}{15}} \approx 1.06$$

(2) To perform these calculations, both the VIPRE and XCOBRA-T code were driven with an average core power oscillation of 30 % of rated power (conservative with respect to the observed 26 %). The core pressure drop was forced to follow the measured pressure drop; this resulted in a peak-to-peak flow oscillation of 26 %. The calculated CPR change was 0.17 for the VIPRE code and 0.18 for the XCOBRA-T code.

For the conditions at the time of the instability, the MCPR at the hot channel (32-33) was 1.946, the OLMCPR was 1.795, and the SLMCPR was 1.07. Therefore, the MCPR during the oscillations was 1.78 (= 1.946 - 0.17), which was less than the OLMCPR but not less than the SLMCPR.

(3) Control rod sequencing procedures at four other BWR sites were reviewed to determine if practices at WNP-2 were abnormal. Discussions with resident inspectors and licensee staff identified some differences in startup strategies; however, most differences could be attributed to different plant design or operating limits. Two major differences between sites include variable speed recirculation pumps, and the restriction that WNP-2 placed on operating recirculation pumps in fast speed. However, procedures from the four other sites consistently followed two operating strategies that were consistent with WNP-2 procedures:

- A qualified reactor engineer could authorize changes to the control rod withdrawal sequence without peer review.

- Neither quantitative guidance nor requirements were specified to control reactor power distributions during startup.

The AIT concluded that the control rod sequencing procedures in use at WNP-2 were consistent with other licensees.

(4) The utility used a training update system to incorporate information from operating events, licensee event reports (LERs), problem evaluation requests (PERs), design changes, and generic information. Coordination with the Operations department was conducted primarily through the Operations Procedures Supervisor, who reported directly to the Operations Manager for development and modification of procedures.

The licensee also described a secondary strategy for developing training material that incorporated information from industry experience and various industry committees, including the BWR Owners' Group. This strategy was an informal process in which information from industry committees was distributed according to a standard distribution list for the specific topic. Personnel receiving the information decided how to act upon it. The March 18, 1992 BWROG letter was distributed according to this list, and went to the Manager, Reactor Engineering, the Manager, Nuclear Engineering, the STA trainer, and the former Supervisor of shift nuclear engineers, who was the SNE/STA on shift at the time of the event.

(5) LAPUR calculations (See Appendix E) indicate that the proposed startup path (case-8 control rod pattern) should result in stable operation. The maximum decay ratio calculated by LAPUR during the proposed startup is 0.3, and it corresponds to the operating point just following the closure of the flow control valve (FCV).

(6) Stability calculations for future startup calculations will be performed by the STAIF code. STAIF is a frequency-domain code with 1-D neutron kinetics and multiple thermohydraulic channels. STAIF validation has not been reviewed by NRC. SNP has proposed to submit validation documentation to the NRC in June 1993. The AIT performed a set of audit calculations of STAIF analyses using the LAPUR code. This audit has resulted in good agreement between the two codes (between 10 % and 20 % mismatch in calculated decay ratios). STAIF calculations for the August 15 event resulted in a decay ratio of 0.86, which is lower by 15 % to 20 % than the actual decay ratio during the event.

(7) These analyses were as follows:

Cycle 8 Fuel Cycle Design Report - March 1992
(Provided the results of fuel cycle design calculations, design core loading, projected control rod patterns, and evaluations of shutdown margin, hot excess, and thermal margins for the core design)

Cycle 8 Reload Analysis - March 1992
(Summarized the results of various safety analyses for the cycle 8 reload including the fuel mechanical design analysis, the thermal hydraulic design analysis, the nuclear design analysis and the anticipated operational occurrences analyses)

Cycle 8 Plant Transient Analysis Report - June 1992
(Provided the results of the fuel vendors evaluation of postulated system transient events during cycle 8 operation)

Cycle 8 Startup and Operations Letter Report - June 1992.
(Provided the neutronics information necessary for cycle 8 startup including the initial estimated critical position and target control rod patterns)

(8) The operators scrammed the reactor at 0300 on August 15, 1992. The NRC resident inspector was called at home at approximately 0315, and was given an initial briefing by the Shift Manager. At 0320 an Unusual Event was declared by the shift manager. Notification of the emergency response team was made at 0327, and the Emergency Operations Facility Communications Center (EOFCC) was notified via facsimile at 0335. These notifications initiated automatic pagers to notify the emergency response team. The EOFCC duty officer then notified the state, local, and other federal agencies by facsimile or telephone. All state, local and appropriate federal agencies were notified within 15 minutes of the declaration. At 0355, the licensee notified the NRC Headquarters Duty Officer via the Emergency Notification System (ENS). The licensee terminated the Unusual Event at 0430 on the basis that the plant was in a stable shutdown condition. The EOFCC was notified that the UE was terminated at 0440. The EOFCC, in turn, made all of the appropriate notifications. The NRC was notified via ENS that the UE was terminated at 0446.

APPENDIX B

Detailed Sequence of Events

The following table gives the sequence of events as reconstructed by the AIT through interviews, calculations, and review of licensee records:

Time (Hours)	Description of Events
2355:00 August 14	Reactor Power held at 34 %, for Turbine Bypass Valve Testing and control rod drive timing testing of six control rods.
0053:14 August 15	A power distribution limits printout (known as "MON Run") was obtained. Core thermal Power was 37.1 percent, and total core flow was 30.5 percent. CMPF was 3.247, MCPR was 1.864, APF was 1.46, the radial peaking factor (RPF) was 1.59, and the T factor was .870.
0256:14	Commenced shutting RRC-FCV-60A. Reactor power was at 36.4 %, and total core flow was at 30.5 %. The reactor was on the 74.9 % rod line. A MON run taken just prior to the valve movement indicated CMPF was 3.782, MCPR was 1.801, APF was 1.59, RPF was 1.63, and the T factor was .746. Xenon reactivity worth was approximately 2.4 % dk/k. Nearly all shaping control rods had been fully withdrawn from the core.
0258:18	Onset of power oscillations from APRM recorder.
0259:03	FCV-60A Fully Shut. The reactor was operating on the 74.2 percent rod line. (33.5 % reactor power, 26 % core flow) Operators noted LPRM downscales randomly lighting across full core display.
0259:49	Rod Block Alarm received (APRM Flow Biased, first of 17 received at two second intervals). At this time APRMs were oscillating between 23 and 47 % power peak-to-peak every two seconds.
0259:59	APRM Upscale Alarm received (Flow Biased, first of 5 received over the next 38 seconds).
0300:28 (est.)	Operators evaluate power oscillations, Shift Manager determined that a reactor scram was required due to power oscillations >10 % peak to peak.
0300:38	Manual Reactor Scram. All control rods verified inserted, and reactor power verified at 0. The turbine tripped, and electrical power supply shifted automatically to the startup transformer from the main turbine generator.

Time (Hours)	Description of Events
0300:44	Reactor Water Level decreased to +13", operators entered Emergency Operating Procedures.
0300:51	Reactor Water Level was at its lowest level (-15") on the wide range level instrumentation.
0301:23	Reactor Water Level increased to +13" via Feedwater Level Control System in Automatic mode, and operators maintain level >+13".
0311	The Shift Manager called the NRC Resident Inspector at home, and briefed the resident inspector on the event.
0320	The Shift Manager declared an Unusual Event.
0337	REA-RIS-19 (Containment LOCA Radiation Monitor) levels increased from 60 cpm at 0300 to a peak of 1500 cpm.
0355	Operators informed NRC Headquarters Operations Officer via ENS of the Unusual Event and Engineered Safety Features Actuation.
0430	Unusual Event terminated based on stable plant conditions.
0747	Reactor coolant sample isotopic analysis revealed normal levels of all isotopes.

The APRM flow-biased Rod Blocks and upscale alarms were received because the setpoint for these trips was approximately 49.5 %, and the peak power during the oscillation event appeared to fall within the instrument tolerance of the APRM flow biased trips (the peak power noted on any of the APRM channels was 48.75). The increase in containment LOCA radiation monitors appeared to be due to the venting of the scram discharge volume to the equipment drains.

APPENDIX C

Stability Calculations for the August 15, 1992 Event

Using the best estimates available for power, flow, and power distribution, LAPUR predicts a corewide decay ratio of 1.05 for the conditions of the oscillations. To achieve this agreement with the data, it was required to model in detail the mixed core conditions present at the time in WNP-2. Tables C.1 and C.2 summarize some of the results from these LAPUR runs. Two conclusions can be reached from the results in Tables C.1 and C.2:

(1) The hot channel was most probably stable, but not with much margin (decay ratio 0.83). A single channel thermohydraulic instability was not likely, but the AIT could not discard it without analyses because of the highly peaked radial power in this event.

(2) Even though the event data shows that the instability was clearly of the in-phase or fundamental mode, the LAPUR analyses indicated that the out-of-phase mode was fairly unstable too. From these LAPUR results, the AIT could determine which of the two modes would likely dominate. Therefore, other startup events with these highly peaked power distributions may result in out-of-phase oscillations.

The AIT performed some sensitivity analyses to identify the root cause of this instability. As stated before, the root cause was the highly peaked radial and axial power distribution. The other contributing factor was the mixed-core characteristics present in WNP-2 at the time of the event. For example, the AIT performed a LAPUR run with exactly the same input conditions, but changing all 9x9-9X fuel for 8x8 fuel. The resulting core and hot-channel decay ratios were reduced by 20 % to 0.87 (core) and 0.67 (hot channel). The AIT performed another LAPUR run similar to the one above, but substituting all 8x8 fuel by 9x9-9X. In this last case, the core decay ratio was 0.95 and the hot channel decay ratio was 0.73. Therefore, the AIT concluded that the mixed core is 10 % worse than a full core of 9x9-9X fuel. Table C.3 summarizes these results.

Based on the LAPUR studies, the AIT performed some more sensitivity studies to define a best-estimate region where WNP-2 should have been unstable under the August 15 startup conditions. This exclusion region is based on the radial and axial power distributions observed on August 15. Figure C.1 shows the LAPUR-calculated unstable region based on the August 15 power distributions. In this figure, the August 15 instability event condition is shown along with the startup path that the reactor operator followed. The AIT concluded from these best-estimate LAPUR calculations that:

(1) The operating point where the oscillations were observed is within the unstable region for the in-phase (or core wide) oscillation mode, and barely outside the out-of-phase (or regional) oscillation mode. Note that these are best-estimate, after-the-fact calculations. If these had been predictive-type calculations, the AIT would have added some conservatism (at least a 20 % factor for the calculated decay ratio) so that it would be predicted that this operating point could have oscillated in either the in-phase or the out-of-phase modes.

(2) The operating point at 36% power and 30% flow, before the flow control valve was closed, was barely stable; therefore, should the ANNA decay ratio monitoring system had been operational, it would have displayed a large (about 0.9) decay ratio before the pump upshift operation was started.

Table C.1 LAPUR-calculated channel decay ratios for nominal 8/15 conditions

Region number	Number Bundles	Relative Power	Relative Flow	Channel Decay Ratio
1	82 9x9-9X	136%	98%	0.46
2	47 9x9-9X	162%	92%	0.61
3	14 9x9-9X	187%	87%	0.83
4	125 8x8	106%	104%	0.04
5	121 8x8	134%	101%	0.18
6	25 8x8	156%	97%	0.41
7	350 mixed	62%	101%	0.0

Table C.2 LAPUR-calculated out-of phase decay ratios for nominal 8/15 conditions

Assuming Eigenvalue separation between first azimuthal and fundamental mode is	First azimuthal (out-of-phase) mode decay ratio
-$0.5	1.07
-$1.0	1.00
-$1.5	0.92

Table C.3 Comparison of stability of mixed core versus single-fuel cores

Core Fuel	Core decay ratio	Hot-channel decay ratio	Out-of-phase decay ratio*
Mixed	1.05	0.83	1.00
8x8	0.87	0.67	0.79
9x9-9X	0.95	0.73	0.89

* Assumes -$1.00 eigenvalue separation

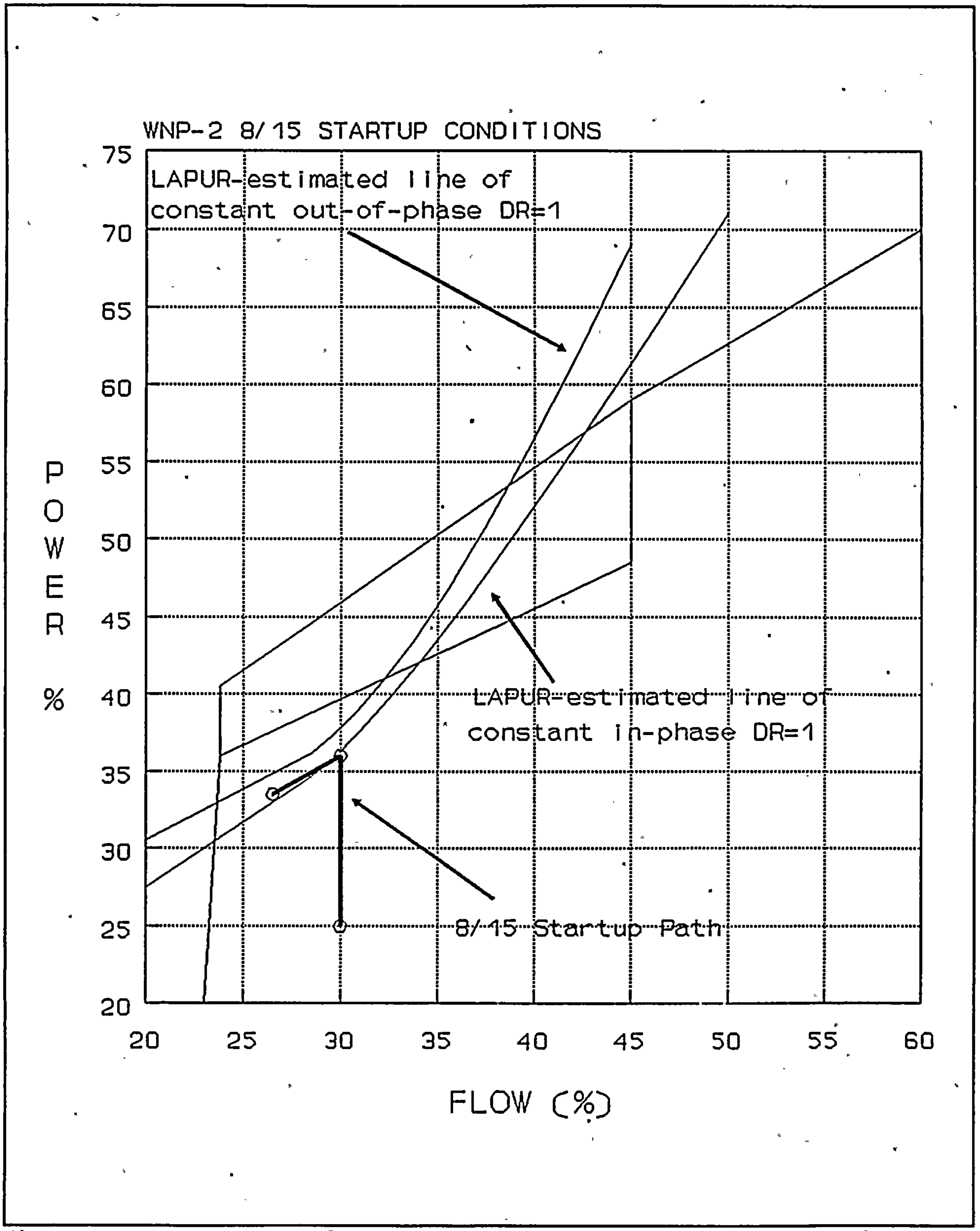

Figure C.1 Best-estimate lines of constant decay ratio=1.0 for actual conditions of 8/15 event, assuming constant power distribution

APPENDIX D

Stability Calculations for the August 30, 1992 Startup

The AIT performed a series of LAPUR calculations to determine the stability of the new startup path proposed by WNP-2. This new startup path was designed to minimize the axial and radial peaking factors by selecting a nonaggressive control rod pattern. Some of the parameters that define this proposed startup are shown in the predicted MON output shown in Figure D.1. Based on these parameters and the predicted full 3-D power distribution supplied by the licensee, the AIT performed a series of LAPUR calculations to determine the stability of the proposed conditions.

The results are summarized in Table D.1 and Figure D.2. The maximum calculated core decay ratio during the proposed startup is 0.3, which indicates a significant margin to instability limits.

Table D.1 LAPUR calculated decay ratios for proposed WNP-2 startup

Power (%)	Flow (%)	Core decay ratio	Out-of-phase decay ratio[1]
30%	30%	0.25	0.08
33%	30%	0.28	0.12
30.3%	26.3%	0.30	0.16

[1] Assumes a -$1.0 eigenvalue separation

Case 8

```
BEST ESTIMATE #7 WITH FCV CLOSED AND FLAT INITIAL GUESS

CORE PERFORMANCE LOG -- SHORT EDIT - PREDICT CALCULATION

   CALCULATION TYPE : NORMAL          CONVERGENCE : TIGHT    SYMMETRY : FULL
   CTP CALCULATION  : HEAT BALANCE    SAVS-92JUN25-162052    W2C8F1 TWO TSSS BOC

STATE CONDITIONS          FLOW RATES    CORE PARAMETERS     NUCLEAR LIMITS     LOCATION
 GMWE   336.40            WT             CMEQ    0.2333     CMPF       3.203   25-26-07
 GMWT   1008.0 (30.3%)    WTSUB          CAEQ    0.1346     CMFLCPR    0.688   41-20
 PR      962.5            WTFLAG         CAOA    0.0445     CMAPRAT    0.329   25-36-07
 DHS      32.00           WFW            CAVF    0.2858     CMFDLRX    0.344   25-26-07
 WT       28.54 (26.3%)   WD             CAPD   14.9098     CMFDLRC    1.134   25-26-07
 PRATIO      1.048                       RWL
 ER          1.052                       COLP   -0.8448     P-PCS      -4.73   25-28-08
 ERATIO      1.004        TARGET 1.048   DPCC    4.7782     P-PCFC     -4.73   25-28-08
 CYCLE EXPOSURE      513.5 MWD/MTU       KEFF    1.0089     P-PFPR     -8.59   25-26-07

 LOCATION            1    2    3    4    5    6    7    8    9   10   11   12
 AXIAL REL POWER  0.31 0.80 1.14 1.34 1.31 1.22 1.14 1.13 1.12 1.03 0.89 0.58
 REGION REL POWER 0.92 1.04 0.92 1.04 1.20 1.04 0.92 1.04 0.92
 RING REL POWER   0.83 1.33 1.05 1.05 1.02 1.26 1.08 0.66
```

************** CONTROL ROD DATA **************

	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	
59					--	--	--	--	--	--	--					59
55				--	--	08	--	00	--	08	--	--				55
51			--	--	18	--	38	--	38	--	18	--	--			51
47		--	--	06	--	00	--	00	--	00	--	06	--	--		47
43	--	--	18	--	42	--	16	--	16	--	42	--	18	--	--	43
39	--	08	--	00	--	00	--	00	--	00	--	00	--	08	--	39
35	--	--	38	--	16	--	38	--	38	--	16	--	38	--	--	35
31	--	00	--	00	--	00	--	04	--	00	--	00	--	00	--	31
27	--	--	38	--	16	--	38	--	38	--	16	--	38	--	--	27
23	--	08	--	00	--	00	--	00	--	00	--	00	--	08	--	23
19	--	--	18	--	42	--	16	--	16	--	42	--	18	--	--	19
15		--	--	06	--	00	--	00	--	00	--	06	--	--		15
11			--	--	18	--	38	--	38	--	18	--	--			11
07				--	--	08	--	00	--	08	--	--				07
03					--	--	--	--	--	--	--					03
	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	

Figure D.1 Proposed conditions for next startup. Two pumps at minimum speed, one flow control valve closed.

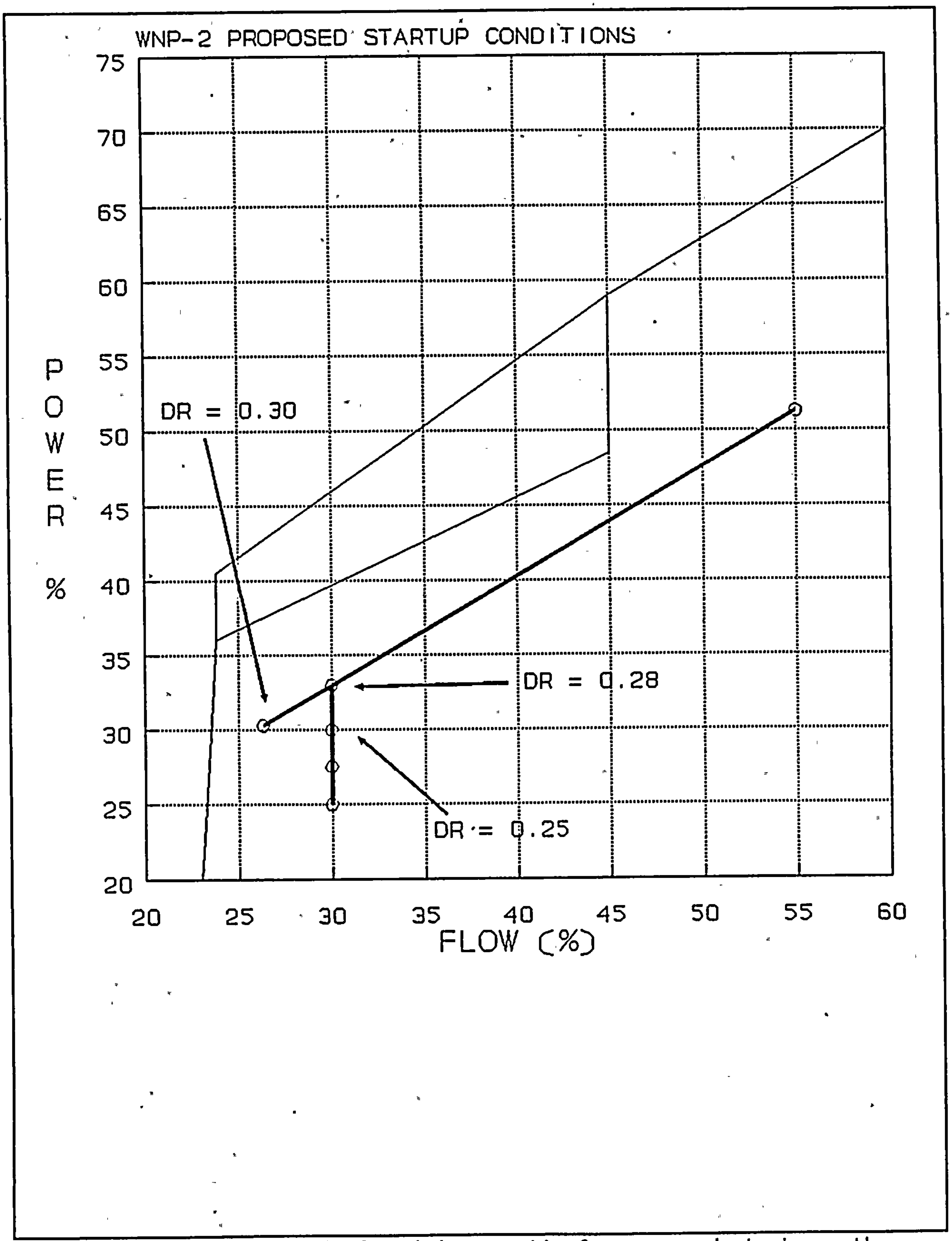

Figure D.2 LAPUR calculated decay ratio for proposed startup path

APPENDIX E

Stability Calculations for the August 2, 1992 Startup

On August 2nd, the WNP-2 plant was started up following a less aggressive rod withdrawal sequence than on the August 15th startup that resulted in unstable power oscillations. This August 2 startup can be used as an example that the Cycle 8 core can be operated safely without startup instabilities if care is taken to minimize the radial and axial peaking factors. Figures E.1 and E.2 provide a comparison of the control rod patterns and limiting parameters for the two startups. As it can be observed, the August 2 startup reached a smaller maximum core peaking factor (3.468 compared to 3.856 for August 15), significantly lower core-average axial peaking factor (1.29 compared to 1.62 for August 15) and lower CPR ratio (0.826 compared to 0.922 for August 15). All these conditions favored the stability of the August 2 startup.

The August 2 operating conditions were reached by not withdrawing 39-30 and symmetric rods (12 notches compared to 28 notches for August 15) as much, and compensating the additional reactivity by pulling more rods all over the core. The August 2 control rod pattern results in a more uniform radial and axial power profiles.

SNP estimated, using the STAIF frequency domain code, that the in-phase-mode decay ratio for the August 2 startup was 0.78 (compared to 1.01 for the August 15 startup). The improvement in decay ratio value is mostly due to control rod pattern and the resulting power distribution. If the August 15 startup had been run with the August 2 control rod pattern, STAIF estimated that the decay ratio would have been 0.68 (the August 15 startup was performed at a lower rod line that the August 2 startup).

WNP-2 WK-9232 92AUG02-09.54.20 224 MWD/MTU PREDICT CALCULATIO[illegible]

CORE PERFORMANCE LOG -- SHORT EDIT - PREDICT CALCULATION

CALCULATION TYPE : NORMAL CONVERGENCE : TIGHT SYMMETRY : FULL
CTP CALCULATION : HEAT BALANCE SAV$-92JUN25-162052 W2C8F1 TWO TSSS [illegible]

STATE CONDITIONS		FLOW RATES		CORE PARAMETERS		NUCLEAR LIMITS		LOCA[illegible]
GMWE	392.66	WT	29.6	CMEQ	0.2934	CMPF	3.468	35-36
GMWT	1194.1 (35.9%)	WTSUB	37.85	CAEQ	0.1562	CMFLCPR	0.826	35-36
PR	963.3	WTFLAG	2	CAQA	0.0527	CMAPRAT	0.422	35-36
DHS	34.80	WFW	4.80	CAVF	0.3273	CMFDLRX	0.441	35-36
WT	29.64 (27.3%)	WD	7.18	CAPD	17.6625	CMFDLRC	1.228	35-36
PRATIO	1.060			RWL	35.7981			
ER	1.052			CDLP	-0.8242	P-PCS	-4.32	17-28
ERATIO	1.004	TARGET	1.048	DPCC	4.8673	P-PCFC	-4.32	17-28
CYCLE EXPOSURE	223.4 MWD/MTU			KEFF	1.0079	P-PFPR	-7.32	35-36

LOCATION	1	2	3	4	5	6	7	8	9	10	11	1[illegible]
AXIAL REL POWER	0.54	1.19	1.29	1.26	1.16	1.07	1.00	0.97	0.99	0.99	0.92	0.[illegible]
REGION REL POWER	0.90	1.09	0.88	1.07	1.28	1.05	0.86	1.06	0.88			
RING REL POWER	0.82	1.37	1.20	1.29	1.10	1.28	1.04	0.47				
APRM GAFS	0.65	0.69	0.68	0.66	0.67	0.66						

--More--

************* CONTROL ROD DATA *************

	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	
59					--	--	00	--	00	--	--					59
55				00	--	16	--	12	--	16	--	00				55
51			--	--	36	--	--	--	--	--	36	--	--			51
47		00	--	08	--	08	--	00	--	08	--	08	--	00		47
43	--	--	36	--	--	--	--	--	--	--	--	--	36	--	--	43
39	--	16	--	08	--	00	--	12	--	00	--	08	--	16	--	39
35	00	--	--	--	--	--	--	--	--	--	--	--	--	--	00	35
31	--	12	--	00	--	12	--	00	--	12	--	00	--	12	--	31
27	00	--	--	--	--	--	--	--	--	--	--	--	--	--	00	27
23	--	16	--	08	--	00	--	12	--	00	--	08	--	16	--	23
19	--	--	36	--	--	--	--	--	--	--	--	--	36	--	--	19
15		00	--	08	--	08	--	00	--	08	--	08	--	00		15
11			--	--	36	--	--	--	--	--	36	--	--			11
07				00	--	16	--	12	--	16	--	00				07
03					--	--	00	--	00	--	--					03
	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	

Fig E.1. Summary of core state for the August 2nd startup conditions with one flow control valve closed and both recirculation pumps at minimum speed.

```
WNP-2            WK-9234 92AUG15-03.00.14   508 MWD/MTU  TRIGR=BACKUP REV=FE

CORE PERFORMANCE LOG -- SHORT EDIT - MON CALCULATION       PPLXBU

   CALCULATION TYPE : NORMAL         CONVERGENCE : TIGHT     SYMMETRY : QUARTE
   CTP CALCULATION  : HEAT BALANCE   SAV$-92JUN25-162052     W2C8F1 TWO TSSS B

STATE CONDITIONS          FLOW RATES       CORE PARAMETERS    NUCLEAR LIMITS        LOCAT
 GMWE   336.40            WT         29.0  CMEQ      0.3394   CMPF           3.859  35-26
 GMWT  1119.0 (33.7%)     WTSUB     38.94  CAEQ      0.1457   CMFLCPR        0.922  35-36
 PR     963.3             WTFLAG        2  CAQA      0.0494   CMAPRAT        0.441  35-26
 DHS    35.81             WFW        4.26  CAVF      0.3450   CMFDLRX        0.460  35-26
 WT     29.02 (26.7%)     WD         7.63  CAPD     16.5517   CMFDLRC        1.367  35-26
 PRATIO     1.216                          RWL      35.8187
 ER         1.052                          CDLP     -0.8448   P-PCS          -3.63  33-18
 ERATIO     1.004         TARGET 1.048     DPCC      4.7954   P-PCFC         -3.63  33-18
 CYCLE EXPOSURE        508.6 MWD/MTU       KEFF      1.0059   P-PFPR         -7.07  35-26

 LOCATION                1    2    3    4    5    6    7    8    9    10   11   1
 AXIAL REL POWER       0.78 1.62 1.60 1.42 1.26 1.14 1.01 0.89 0.79 0.66 0.51 0.
 REGION REL POWER      0.87 1.06 0.86 1.07 1.50 1.06 0.84 1.01 0.87
 RING REL POWER        0.96 1.61 1.41 1.37 1.09 1.26 0.97 0.42
 APRM GAFS             0.95 0.96 0.94 1.00 0.93 0.94
--More--

             ************* CONTROL ROD DATA *************

       02   06   10   14   18   22   26   30   34   38   42   46   50   54   58
  59                       --   --   00   --   00   --   --                       59
  55                  00   --   00   --   00   --   00   --   00                  55
  51             --   --   --   --   --   --   --   --   --   --   --             51
  47        00   --   00   --   00   --   00   --   00   --   00   --   00        47
  43   --   --   --   --   --   --   --   --   --   --   --   --   --   --   --   43
  39   --   16   --   00   --   00   --   28   --   00   --   00   --   16   --   39
  35   00   --   --   --   --   --   --   --   --   --   --   --   --   --   00   35
  31   --   00   --   00   --   28   --   00   --   28   --   00   --   00   --   31
  27   00   --   --   --   --   --   --   --   --   --   --   --   --   --   00   27
  23   --   16   --   00   --   00   --   28   --   00   --   00   --   16   --   23
  19   --   --   --   --   --   --   --   --   --   --   --   --   --   --   --   19
  15        00   --   00   --   00   --   00   --   00   --   00   --   00        15
  11             --   --   --   --   --   --   --   --   --   --   --             11
  07                  00   --   00   --   00   --   00   --   00                  07
  03                       --   --   00   --   00   --   --                       03
       02   06   10   14   18   22   26   30   34   38   42   46   50   54   58

      CONTROL ROD SEQUENCE :   A-2      CONTROL ROD DENSITY :   0.267
      CONTROL RODS SYMMETRIC
```

Figure E-2. Summary of core state for the August 15th startup conditions with one flow control valve closed and both recirculation pumps at minimum speed.

APPENDIX F

Review of Previous Cycle 8 Startup Reactivity Anomalies

On July 4, 1992, the licensee conducted two startups. During both of these startups the core was critical on a rod significantly before the expected rod. On July 4, 1992, which was the initial post-refueling startup, the reactor went critical ten rods before the expected critical position in one half of the core. Eight of those ten rods were lower worth rods. The Manager, Reactor Engineering, and personnel from the fuel vendor, were present to observe this startup. Based on the judgement of the personnel present, the decision was made that no abnormal conditions existed. The startup continued based on the recommendation from the Manager, Reactor Engineering.

Based on reload analyses calculations SNP had predicted that criticality would be reached when pulling rod 18-15, which is 12 rods behind rod 26-07 (the actual critical rod) in the startup sequence (see Fig. F.1). SNP now estimates that the cold critical calculations were in error by approximately 0.5 % ΔK/K (or 5 mK); they attribute the error in part to the fact that Cycle 7 was a very short cycle and the residual gadolinium concentration was hard to predict. SNP presented some historical data (see Fig. F.2) which indicated that most of their cold-critical k-effective calculations lie within a band that is 1.0 % ΔK/K wide. Therefore, the licensee concluded that the 0.5 % ΔK/K error was not unusual. The TS permitted a maximum cold-critical k-effective error of 1.0 % ΔK/K.

A secondary effect of mispredicting the rod at which criticality occurs is that rod 26-07 is a high worth rod that, when withdrawn, results in a large flux perturbation. This high worth resulted in the flux tilt that was observed during the startup. As a general rule, it is preferable that criticality be reached with a low-worth rod to minimize flux tilts.

The Shift Manager submitted a Problem Evaluation Request (PER) to have the problem formally evaluated. The Explanation and Proposed Resolution, provided by the Manager, Reactor Engineering, provided several factors that contributed to what was termed "a situation of uncommon flux distribution during the initial critical." The factors included a high worth rod being pulled in proximity to SRMs and IRMs, which made the flux appear higher in the vicinity of the rod being pulled. The PER further explained that control room personnel should expect this type of response whenever high worth rods are pulled resulting in an asymmetric rod pattern. If this situation created difficulty to Operations personnel, then banking high worth rods together with their symmetric counterparts was suggested within the constraints of the Rod Sequence Control System.

While this event was not the focus of the AIT, consideration of this event as it related to the core power instability event was specified in the AIT charter. The AIT could not find any technical nexus between the early criticalities and the power oscillation events. The AIT noted that the Shift Manager was concerned enough to generate the PER, and should be commended for pursuing resolution of this anomaly.

The AIT was concerned, however, with the licensee's position that early criticalities, even with high worth rods, were to be expected. The focus of the PER-provided solution was on the flux tilt, and not on the early criticality. Furthermore, rather than direct that the shift staff stop and ensure a reactivity anomaly is well understood, the Reactor Engineering Department provided a means to get around flux anomalies by pulling banked rods. How this would correct for the early criticality was not explained.

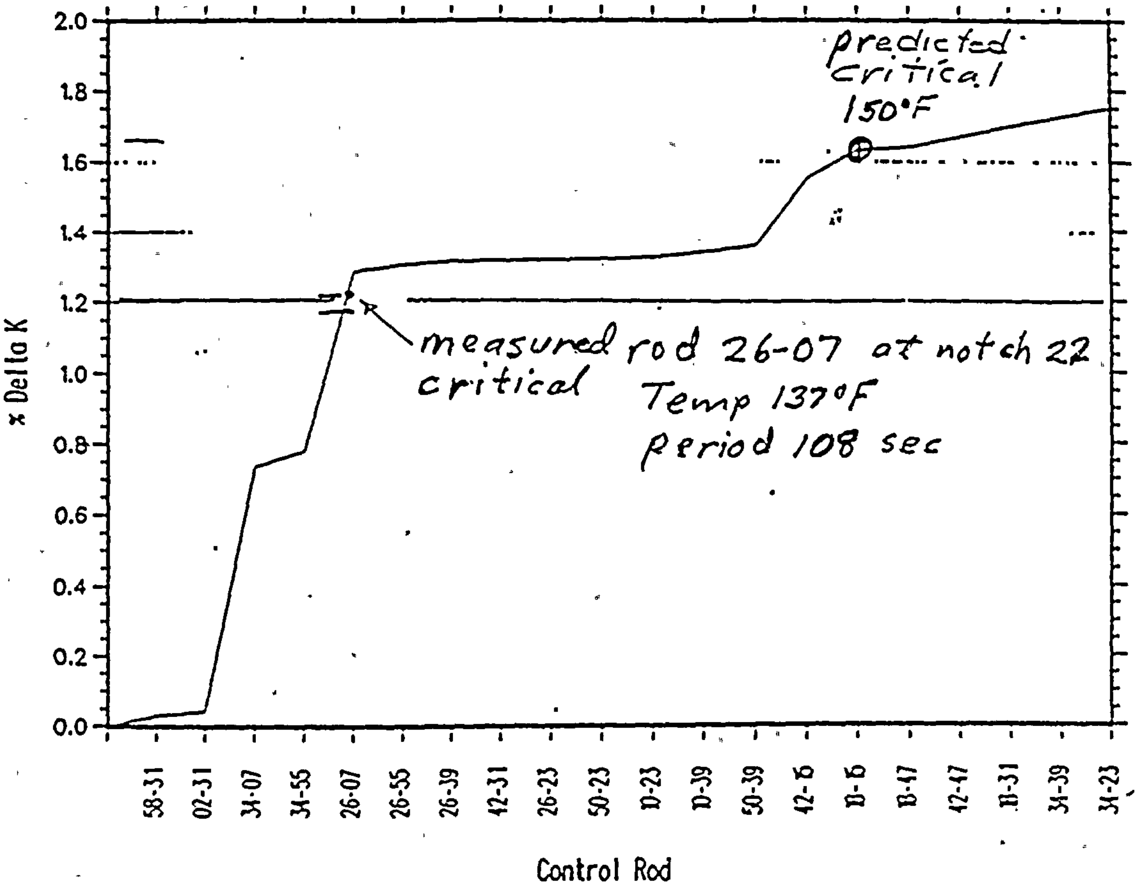

Fig. F.1. Control rod worths for the selected startup withdrawal sequence. Criticality was predicted for rod 18-15, but was reached at rod 26-07, a 0.5% ΔK/K error.

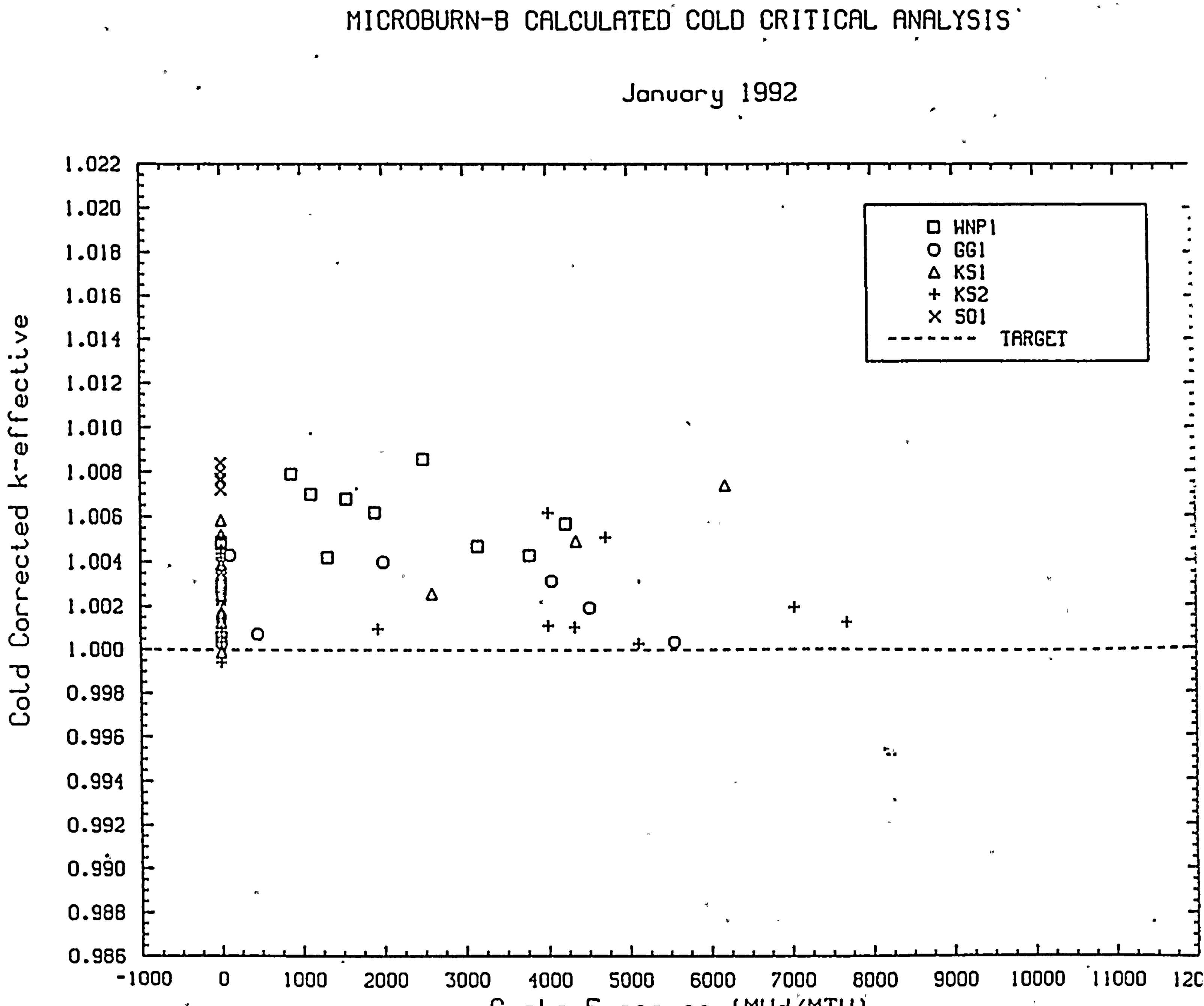

Fig. F.2. Historical cold-corrected k-effective values predicted by SNF, showing a 10% ΔK/K bounding error band.

APPENDIX G

Survey of Rod Sequence Practices at Other BWRs

LA SALLE:

Procedures reviewed:
- LAP-100-13, "Control Rod Sequence Package Preparation, Review, and Implementation"
- LTP-1600-2, "Guidelines For Control Rod Sequence Development"

A Qualified Reactor Engineer (QRE) may authorize changes to the sequence, and documents approval on LAP-100-13 Attachment F.

A new sequence, or major revision (not defined) must be approved by two QREs, including the lead QRE or designee.

No prescriptive guidance regarding peaking factors or power distributions during startup were identified.

SUSQUEHANNA:

Procedures reviewed:
- AD-QA-138, "Control of Core Reactivity Changes At SSES"
- RE-2TP-013, "Power Ascension and Shaping Using Control Rods"

A Qualified Reactor Engineer (QRE) may authorize changes to the sequence, and documents approval on forms AD-QA-138-10 and AD-QA-138-2 (page 13). Procedure specifically identifies this change method for setting rod pattern on startup.

A new sequence must be approved by two QREs, including the lead QRE or designee.

No prescriptive guidance regarding peaking factors or power distributions during startup were identified.

HOPE CREEK

Procedures reviewed:
- HC.RE-IO.ZZ-001(Q), "Core Operations Performance"
- RE.FM.ZZ-001(Q), "Guidelines for Control Rod Movement - Power Operation"

A Qualified Reactor Engineer (QRE) may authorize changes to the sequence, and documents approval on approved forms. Prescribed control rod patterns are followed to the 60% rod line, with 55% total core flow (variable speed recirculation pumps). Flux shaping is then performed - only limits are Technical Specification thermal limits.

No prescriptive guidance regarding peaking factors or power distributions during startup were identified.

GRAND GULF:

Procedures reviewed:
17-S-02-400, "Control Rod Sequences and Movement Control"

A Qualified Reactor Engineer (QRE) may authorize changes to the sequence, and documents approval on a Movement Tracking Sheet (Attachment II or III). Page 9 gives the QRE specific (broad) authority to modify or amend the sequence.

A new sequence, or major revision (not defined) must be approved by two QREs, including the Reactor Engineering Superintendent or designee.

No prescriptive guidance regarding peaking factors or power distributions during startup were identified.

DUANE ARNOLD -

Procedures reviewed: RCP-DI-#5

A Qualified Reactor Engineer (QRE) must revise the entire pull sheet to perform any sequence modification. This revision is then approved by reactor engineering group leader, and operations. Any major events (startup, sequence exchange, etc.) are discussed with their GE refueling representative.).

Duane Arnold has variable speed recirculation pumps, and typically does not pass above the 70% rod line while near the flow instability region.

No prescriptive guidance regarding peaking factors or power distributions during startup were identified.

APPENDIX H
AUGMENTED INSPECTION TEAM CHARTER

WASHINGTON NUCLEAR PLANT 2
POWER OSCILLATIONS ON AUGUST 15, 1992

The Augmented Inspection Team (AIT) is to perform an inspection to accomplish the following:

1. Develop a complete sequence of events and description of the power oscillations which occurred on August 15, 1992. In addition, develop a sequence of events and description of the plant reactivity anomalies which occurred during the plant startup from the most recent refueling outage. Include in the sequence of events operator actions, decision making and communication leading up to the decision to trip the reactor.

2. Identify those equipment failures human performance errors, procedural deficiencies, and quality assurance deficiencies that contributed to these events. In making this analysis, specifically include the following:

   a. Determine what procedures the licensee had implemented to avoid the mitigate power instability, and assess the adequacy of these procedures.

   b. Review the core fuel loading and rod patterns in effect during the event. Review the most recent core reload safety analysis to determine whether the predicated areas of flow instability were properly calculated. Assess whether these areas were contributing causes for the events.

   c. Review the operating status of the recirculation flow control system. Assess whether the operation of the components in this system were contributing causes foe the events.

   d. Review the operating status of the turbine control system. Assess whether the operation of the components in this system were contributing causes for the events.

   e. Assess the operating crew's performance during and subsequent to each event. Ascertain the control room organization and staffing level during the event and the role of the STA before, during and after the event.

3. Evaluate the effectiveness of the licensee's investigation of these events.

4. Determine the root causes of the power oscillation event, from equipment, personnel, and organizational perspectives.

5. Assess the adequacy of the licensee's notification of other agencies of both of these events.

Inspection Procedure 93800, "Augmented Inspection Team Implementing Procedure" and Manual Chapter 0325, "Augmented Inspection Team" provide additional administrative guidance with will be used by the AIT.

APPENDIX I

1. PERSONS CONTACTED

Washington Public Power Supply System (WPPSS)

D. W. Mazur, Managing Director
*A. L. Oxsen, Deputy Managing Director
*J. V. Parrish, Assistant Managing Director, Operations
*J. W. Baker, WNP-2 Plant Manager
*C. M. Powers, Director, Engineering
*J. C. Gearhart, Director, Quality Assurance
*G. C. Sorensen, Regulatory Program Manager
D. L. Larkin, Engineering Services Manager
D. L. Whitcomb, Nuclear Engineering Manager
*L. T. Harrold, Assistant Plant Manager
W. D. Shaeffer, Operations Manager
R. L. Webring, Technical Manager
W. H. Sawyer, Shift Manager, Operations
L. L. Grumme, Manager, Nuclear Safety Assurance
S. L. Washington, Manager, Nuclear Safety Engineering
*R. L. Romanelli, Manager, Communications
*J. C. Britton, Public Affairs Officer
W. A. Estes, Control Room Supervisor
D. J. Strote, Control Room Supervisor
J. E. Rhoads, Manager, Operations Event Analysis and Resolution
S. L. McKay, Manager, Licensed Operator Training
C. J. Halbfoster, Chemistry Manager
G. H. Wooley, Procurement QA Manager
D. W. Merhar, Operations Procedures Supervisor
D. K. Atkinson, Reactor Systems Supervisor
R. O. Vosburgh, Safety Analysis Supervisor
S. R. Kirkendall, Plant Support Engineering Supervisor
D. L. Moore, Control Room Operator
G. A. Westgard, Control Room Operator
D. D. Hughes, Control Room Operator
R. H. Torres, Principle Engineer, Reactor Engineering
R. J. Talbert, Station Nuclear Engineer, Shift Engineering
D. R. Skeen, Principle Core Analyst Engineer, Fuels
G. C. DeBatista, Principle Procurement QA Engineer
R. M. Simons, Principle Procurement QA Engineer
B. R. Nowack, Operating Experience Engineer
J. J. Muth, Engineer, Operations Event Analysis and Resolution
I. Jenkins, System Engineer
G. J. Freeman, System Engineer

Siemens Nuclear Fuel Corporation

J. Morgan, Vice President, Engineering
L. Federico, Manager, BWR Fuel Engineering (BWRFE)
R. Copeland, Manager, Reload Licensing
C. Volmer, Manager, QA
A. Reparaz, Manager, Fuel Design

J. Ingham, WNP-2 Reload Team Leader, BWRFE
D. Pruitt, Engineer, BWRFE
P. Wimpy, Senior Engineer
R. Nelson, Senior QA Engineer
S. Jones, Senior Engineer
J. Maryott, Staff Engineer

General Electric (GE)

H. C. Pfeffler, Licensing Project Manager
C. R. Boznak, Site Service Manager
D. A. Salmon, Senior Engineer

Bonneville Power Administration (BPA)

*J. R. Lewis, Director, Division of Nuclear Projects
A. J. Rapacz, Project Representative

Institute of Nuclear Power Operations (INPO)

P. Huffmeier, Events Analysis

Nuclear Regulatory Commission

N. Conicello, Project Manager, NRR
J. Wechedburger, Regional Coordinator and Policy Analyst, OEDO
N. Hunnemuller, Operator License Examiner, NRR
M. Peck, Acting Resident Inspector, LaSalle Station

Others

*E. Smith, Tri City Herald

* Attended AIT exit meeting on August 29, 1992.

APPENDIX J

Shift Staffing During the Event

The WNP-2 control room was staffed by two licensed senior reactor operators, the Shift Manager (SM) and the Control Room Supervisor (CRS); three licensed reactor operators, known as Control Room Operators (CRO); and a Shift Technical Advisor (STA). This manning met the staffing level required by the Technical Specifications (TS). The STA was certified for his position by the licensee. The operating crew was performing the following duties:

POSITION	FUNCTIONAL RESPONSIBILITIES
SM	The SM came on watch at 2230, and was responsible for providing the operating crew overall direction and control. He was in the control room during the event near the CRO's desk.
CRS	The CRS came on watch at 0230, and was responsible for directly supervising the plant startup. He was in the control room near panel P-603 (which contains the full core display and control rod selectors) during the event.
CRO#1	CRO#1 came on watch at about 1930, and was operating the reactor recirculation system flow control valve (FCV) just prior to the event.
CRO#2	CRO#2 came on watch at 0230, and was coordinating containment inerting with nitrogen from a control room back panel. After the reactor scram, he assisted the other operators with the plant shut down.
CRO#3	CRO#3 came on watch at 1830, and had been performing control rod manipulations at panel P-603 before the event. During and after the event, he monitored balance of plant systems, including the feedwater and condensate systems.
STA/SNE	The STA came on shift at 2230. The STA was also qualified as a Shift Nuclear Engineer (SNE), and is responsible for monitoring and controlling the reactor's neutron flux profile.

APPENDIX K

Partial List of Procedures and Correspondence Reviewed

1. PPM 3.1.2, "Reactor Cold Startup"

2. PPM 9.3.9, "Plant Power Maneuvering"

3. PPM 4.12.4.7, "Unintentional Entry Into Region of Potential Core Power Instabilities"

4. PPM 5.1.1, "RPV Control (Non-ATWS)"

5. PPM 3.3.1, "Scram Recovery"

6. Letter dated September 12, 1988. The licensee's response to NRCB 88-07.

7. Letter dated March 3, 1989. The licensee's response to NRCB 88-07 Supplement 1.

8. NRC Inspection Reports 50-397/88-37 and 89-11, which included review and closeout of the licensee's responses to NRCB 88-07 and NRCB 88-07 Supplement 1.

9. Administrative Procedure 1.10.4, "External Operating Experience Review," which contained the licensee's formal program for review of external operating experience.

APPENDIX L

Definitions of Core Physics Parameters

$$\text{CPR} = \frac{\text{fuel bundle power that would cause departure from nucleate boiling (DNB)}}{\text{hottest operating fuel bundle power}}$$

$$\text{CMFLCPR} = \frac{\text{operating limit CPR}}{\text{actual minimum operating CPR}}$$

$$\text{CMPF} = \frac{\text{hottest fuel pin power}}{\text{average fuel pin power}}$$

or

= (axial x radial x pin) peaking factors at the location of the hottest pin