LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
MANUAL REACTOR SCRAM DUE TO CORE INSTABILITY

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBERS(S)
0 8	1 5	9 2	9 2	-- 0 3 7	-- 0 1	1 1	0 2	9 2		0 5 0 0 0
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OPERATING MODE (9)	X	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	3 3 %	[ ] 20.402(b)	[ ] 20.405(C)	[X] 50.73(a)(2)(iv)	[ ] 77.71(b)
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		[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
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		[ ] 20.405(a)(1)(iv)	[X] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
		[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
C. L. Fies, Compliance Engineer	AREA CODE 5 0 9 3 7 7 - 4 1 4 7

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
[ ] YES (If yes, complete EXPECTED SUBMISSION DATE)	[X] NO				

ABSTRACT (16)

At 0301 hours on August 15, 1992 the reactor was manually scrammed due to indications of core instability. Plant control room operators noted Average Power Range Monitors (APRMs) swinging between 25 and 45 percent power with numerous Local Power Range Monitor (LPRM) downscale indications. The plant was scrammed within 80 seconds after the oscillations were detected.

The root cause for this oscillation event has been determined to be unanticipated interaction of operating conditions and components. The primary reason for the oscillation was the skewed radial and axial power distribution in the reactor. The Supply System staff and vendor personnel failed to identify the extent of the tendency for this core design to become unstable under certain operating practices. The core currently consists of about 26% 9x9 design fuel bundles and 74% 8x8's. The 9x9's have a higher pressure drop compared with 8x8 fuel. In addition, the 9x9 fuel assemblies were operated at a higher power relative to the 8x8's. These conditions led the core to be susceptible to power oscillations. When the first Recirculation Flow Control Valve was closed, in preparation for the Recirculation Pump shift to high speed, thermal-hydraulic instability was initiated and self-limiting power oscillations followed. The event was terminated by a manual reactor scram.

Short term corrective action was taken to revise operating strategies and related procedures to preclude identified precursor conditions and increase precautions and controls during operation at low power levels. Long term design process, operating, and management corrective actions are also being pursued.

Plant Conditions

Power Level - 33%
Plant Mode - 1

Event Description

At 0301 hours on August 15, 1992 the reactor was manually scrammed due to indications of core instability. Plant control room operators noted Average Power Range Monitors (APRMs) swinging between 25 and 45 percent power with numerous Local Power Range Monitor (LPRM) downscale indications in all regions of the full core display. Plant operators manually scrammed the reactor as required by plant procedures and training. An Unusual Event was declared at 0320 hours in response to the power oscillations and plant shutdown.

Prior to the event, the unit was reduced in power from 100% to about 5% power starting at 1855 hours on August 13, 1992 due to high unidentified drywell leakage. A drywell entry was made to locate and isolate the source of the leak. The Reactor Bottom Head Drain Valve Bypass (RWCU-V-103) was found with failed packing and the valve was back seated to stop the leakage.

At 1710 hours on August 14, 1992 the operators commenced returning the unit to 100% power with rod pulls. Since this was approximately (22) hours after 100% power operations, Xenon levels were significantly higher than 100% equilibrium Xenon levels as well as equilibrium values for 5-7% power. This allowed the Station Nuclear Engineer to establish a rod pattern highly beneficial to plant maneuvering after the Recirculation Pump Shift to high speed (60 Hz) but which had high radial and axial peaking and an axial profile peaked in the bottom of the core. The Advanced Nuclear Noise Analysis (ANNA) Monitoring System, WNP-2's stability monitoring system, was not turned on as the plant was operating outside the region for which Technical Specifications required ANNA to be in operation. The unit attained 34 percent power and 30% flow at about 2355 hours on August 14, 1992. The unit was held in this condition until 0258 hours on August 15, 1992 to support Turbine Bypass Valve testing and Control Rod Drive timing. At 0258 hours, power level was increased to about 36.5 percent as required by plant procedures to provide adequate flow margin above the Recirculation Pump Cavitation Interlock. The control room operators then closed RRC-FCV-60A in preparation for recirculation pump 1A shift to 60 Hz speed. At 0259 hours, rod out block alarms and LPRM down scale alarms were received. The operators noted that the APRMs were oscillating between 25% and 45% power. The Shift Manager directed a Manual Scram in accordance with PPM 4.12.4.7 "Unintentional Entry into Region of Potential Core Instability", at 0301 hours. All rods fully entered the core upon the scram. The shutdown of the unit to a cold shutdown condition went normally.

Immediate Corrective Action

At 0415 hours on August 15, 1992, preparation began to place the reactor in cold shutdown and the unusual event was terminated at 0430 hours. Later on August 15, a root cause team was appointed and an analysis plan was formulated to investigate the cause of the event, assist the NRC Augmented Inspection Team (AIT) and recommend corrective actions. The Supply System root cause team was supplemented by experts from Siemens Nuclear Power, General Electric, and INPO.

Further Evaluation, Root Cause, and Corrective Actions

A. Further Evaluation

1. This event is being reported per the requirements of 10CFR50.73(a)(2)(iv) as "....any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)." This event is also reportable under 10CFR50.73(a)(2)(ii)(B) as "....a condition that was outside the design basis of the plant" and is reportable as a Technical Specification required shutdown under 10CFR50(a)(2)(i)(A) since the reactor was scrammed to counter the instability, as directed by the action statements 3.2.6 and 3.0.3. Finally, this event is reportable under 10CFR50.73(a)(2)(ii)(A) as an "....unanalyzed condition that significantly compromised plant safety." A 50.72 report was made to the NRC at 0327 hours on August 15, 1992.

2. Boiling Water Reactor (BWR) cores typically operate with the presence of global neutron flux noise. This noise can be characterized as oscillations having a dominant frequency of 0.3 to 0.5 Hertz and a maximum peak to peak amplitude of five to eight percent of rated flux. They are attributed to flow and boiling in the core, are stable due to the negative void reactivity feedback characteristics (i.e., power increases, voids increase, reactivity decreases, power decreases) and are normally of no consequence to safety, fuel performance or plant operation. Power oscillations of concern are driven by thermal-hydraulic instabilities that occur under relative conditions of high power and low flow. Under these conditions, oscillatory voiding and reflood conditions occur in the impacted fuel assemblies. When this thermal hydraulic instability is combined with the neutronic feedback mechanisms, the observed core oscillations occur.

3. Further evaluation found that radial and axial flux profiles established in the core and the core and fuel assembly design, combined to cause thermal-hydraulic induced power oscillations. The rod pattern selected caused the 9x9 bundles to be operating at higher power levels than the 8x8 fuel resulting in significantly reduced flow due to 1) higher pressure drop across the 9x9 bundle, and 2) higher power level per bundle causing increased flow resistance. Specifically, the 9x9 bundles in the core have a greater hydraulic pressure drop across them as compared to the 8x8 design. Currently in Cycle 8, WNP-2 has about 26% of its core with 9x9 design and the remaining 74% of the core is 8x8 design including 12 LFAs. As can be seen from the attached Figure 1, a significant

number of assemblies were operating at high power and relatively low flow. Thus, a number of individual assemblies were actually operating in the region where instabilities can occur. Enough individual assemblies reached the point of hydraulic instability and the overall core flux oscillations followed.

4. Additional investigations examined events which led the Station Nuclear Engineer (SNE) and operators to place the core in this condition. The high Xenon level allowed the SNE to proceed to the point at which additional shaping rods were withdrawn from the core resulting in a bottom peaked axial power distribution. Also, it allowed the SNE to establish a radial and axial core flux profile to optimize conditions after the pump shift. Specifically, the Nuclear Engineer set a radial flux profile with a Critical Power Ratio (CPR) of 1.812 with a limit of 1.728 giving a Fraction of CPR of .953 for bundle 35-36. The Total Peaking Factor (TPF) was 3.782 for location 35-26-3. The time for which these values were calculated for was 0257, 3 minutes before the power oscillations and just prior to closing the Flow Control Valves. The BWROG guidelines indicate that these factors are contributors to power oscillations when in low flow and high power conditions.

5. At the time of the event one feedwater heater was out of service. This resulted in slightly lower feedwater temperature which did not contribute to the instability.

6. The time spent at relatively high power and low flow (i.e., 34% power, 30% flow) for CRD timing and bypass valve testing increased the chances of a power oscillation event occurring. Since the plant was operating in an area below the previously known region of potential instability there were no restrictions applied.

7. Further evaluation of the activities leading up to the event identified a Technical Specification violation. Technical Specification 3.2.2, Power Distribution Limits/APRM Setpoints, is applicable in Operational Condition 1 when thermal power is greater than or equal to 25 percent of rated. The purpose of this Technical Specification at less than full power conditions is to provide added protection against a highly peaked power distribution by temporarily adjusting the APRM setpoints to a more conservative value. This is done by evaluating Power Distribution data and determining the "T" factor used to adjust the setpoints. Technical Specification 4.0.4 states that entry into an Operational Condition shall not be made unless the Surveillance Requirements associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. Surveillance requirement 4.2.2.c states the surveillance is to be performed "...Initially and at least once per 12 hours when the reactor is operating with Maximum Faction of Limiting Power Density (MFLPD) greater than or equal to the Fraction of Rated Thermal Power (FRTP)." During the power ascension associated with this event the plant reached 25 percent power at 2320 hours on August 15, 1992. At that time, the "T" factor was approximately 0.85. A further decrease occurred between 0100 and 0300 hours with the "T" factor at the time of the event being 0.73. Plant Procedure PPM 7.4.2.1, Power Distribution Limits implements the requirements of this

Technical Specification. The procedure required the "T" factor to be evaluated and the APRM setpoints to be adjusted every twelve hours when "T" is less than one. This was a violation of the Technical Specifications and is reportable under 50.73(a)(2)(i)(B).

8. Further investigation discovered a design problem with the Advanced Neutron Noise Analysis (ANNA) Monitoring System which was installed in 1989. This system monitors the signal of six APRMs and 18 LPRMs. Its purpose is to provide advance warning of both global (in-phase) and regional (out-of-phase) oscillations. The ANNA Monitoring System is governed by Technical Specification 3/4.2.7 and 3/4.2.8, Stability Monitoring. The action statements of these Technical Specifications requires a decrease in thermal power or an increase in core flow within 15 minutes if the decay ratio measured by ANNA is greater than 0.75. The decay ratio is the ratio between the amplitude of two consecutive peaks in the neutron signal. The decay ratio is less than one for a stable system. A decay ratio of 0.75 is selected as a decay ratio limit for operator response such that sufficient margin to an instability occurrence is maintained. The natural frequency of a BWR is approximately 0.3 to 0.5 Hertz. The ANNA Monitoring System monitors between 0.2 and 0.7 Hertz. The LPRM/APRM signals normally have 0.3 Hertz low pass filters to remove noise from the signal used for normal power monitoring. However, this same signal was used by the ANNA monitoring system which, on subsequent design review, needed a 5 Hertz filter to give accurate readings. The use of 0.3 Hertz filter on the input to ANNA resulted in non-conservative values of decay ratio in the region of interest. For example, calculations by Siemens Power Corporation for data associated with this event show that with a 0.3 Hertz filter ANNA would calculate a decay ratio of 0.62 while the decay ratio calculated utilizing unfiltered signals would be 0.89. Thus, the design of the ANNA Monitoring System and its associated signal conditioning did not provide for accurate decay ratio determinations. An additional problem was discovered with Surveillance Procedure PPM 7.4.2.7.3, Core Stability Monitoring. This procedure did not reflect the fact that the input to ANNA was filtered. As a result, the peak to peak amplitude reflected in the procedure was non-conservative. This condition is reportable under 10CFR50.73(a)(2)(ii)(B).

B. <u>Root Cause</u>

This event was analyzed by Plant staff with representatives from General Electric, Institute for Nuclear Power Operation (INPO) and Siemens Power Corporation (Fuel Vendor). The primary root cause for this oscillation event has been determined to be unanticipated interaction of operating conditions and components. The Supply System staff and vendor personnel failed to identify the extent of the tendency for this core design to become unstable under certain operating practices. In addition to the primary root cause, there were several contributing root causes.

1. The first root cause involves Plant/System Operation since the effects of changing operating parameters were not fully evaluated in that:

   a. The SNE selected a start up rod pattern with characteristics of aggressive critical power ratios and high radial peaking. This was standard operating practice to minimize stress on the Reactor Recirculation pumps and the fuel. Maximizing the amount of control rod movement during low speed recirculation pump operation minimizes the amount of rod movement required after the shift to high speed. With the control rods set to support power increases principally by increasing flow, the number of high power changes associated with rod movement is significantly reduced. This results in less local fuel stress at higher power levels. In addition, recirculation flow and power could then be increased quickly minimizing the time spent at high speed with the flow control valves at the minimum position when the pump is subject to increased vibration.

   b. Past start ups and operating regimes during cycle 8 had not disclosed problems with similar patterns.

   c. The plant stability monitor (ANNA) was not employed to provide early warning of the potential changes in core instability. This was, in part, due to past experience and a confirmation that the existing core exclusion region versus actual and planned plant conditions for this start up were acceptable.

Thus, the SNE failed to consider the need to be conservative in his rod pattern selection and use of monitoring tools for conditions that could promote and predict core stability.

2. The second root cause was design related. Specifically, there was inadequate independent review of design changes in that:

   a. The Supply System design review process for the mixed core consisting of 9x9 and 8x8 fuel assemblies failed to discern the impact the differences in hydraulic resistance of the fuel assemblies would have on the core's susceptibility to instabilities outside existing instability regions defined by Technical Specification 3.2.7 and current BWROG guidance.

   b. Design review included assurances of conformance to license requirements, but did not discern that core stability licensing analyses did not consider the effects of high peaking on core stability at operating conditions which existed at conditions other than the licensing basis.

Thus, the design review program responsible for setting limits on the plant failed to adjust the rod pattern and peaking conditions to assure core stability.

3. The third root cause was analysis deficiencies in that:

   a. The fuel designer did not perform sensitivity analyses for core instability at reactor conditions other than those required to perform reload licensing analyses.

   b. Current licensing methodologies do not require these sensitivity analysis on start up power distributions.

   c. Computer models to analyze cores to this level of sophistication are in development and not licensed for use.

   d. The designer believed the hydraulic differences between 8x8 and advanced 9x9 would be offset by the void coefficient and would not contribute to the likelihood of instability under nominal start up conditions.

   Thus, the fuel designers failed to identify that this core was less stable and did not provide recommendations for compensating through conservative operating conditions.

4. A contributing causal factor was management methods. A Management Oversight and Risk Tree (MORT) analysis has been completed for this event. This analysis found significant weaknesses in the barriers management has established to prevent the event and in the management controls in place for the design and operation of the core. Management acknowledges it should have responded to prior industry information and critically questioned the design oversight and operating philosophy to minimize the potential for core power oscillations. Specific findings resulted from this review included the following:

   a. Management's response to the Implementation Guidance for Stability Interim Corrective Actions issued on March 18, 1992 was weak. Training was provided to STAs, SNEs and Plant Operators but procedures were not updated.

   b. Management Policy allowed too much flexibility for the SNE/STA to determine the core flux profile.

   c. Management methods used to review the reload design did not ask penetrating questions in the area of core stability.

   d. Management decisions and reasoning for reanalysis and acceptance of a lower feedwater interlock value were not well communicated. Consequently, procedures were inconsistently amended and SNE operating strategies were not appropriately influenced to take advantages of the lower feedwater value.

5. The root cause for the Technical Specification violation associated with the "T" factors for the APRM setpoints was less than adequate procedures. The Technical Specification requirements were not adequately written into the procedure requiring the adjustment prior to 25 percent power. A contributing root cause was management methods which failed to recognize and take corrective action for the non-compliant condition created by Technical Specification Surveillance 7.4.2.1.

6. The root cause for the problem with the filtered input to ANNA was a design configuration and analysis deficiency. The interface between the existing plant hardware and the ANNA hardware received a less than adequate review and analysis. A contributing cause was an inadequate review and test of the design change to assure operability after installation.

B. <u>Further Corrective Action</u>

The following short term corrective actions have been completed.

1. In order to maintain assembly power to flow ratio as low as possible, as well as maintaining radial peaking as low as possible, procedures were revised to require Critical Power Ratio (CPR) greater than 2.2 between 25 percent power and 50 percent core flow. Core total peaking factor will be maintained less than 3.4 prior to pump shift. These are initial parametric values and will be reevaluated for each cycle during the transition from a mixed 8X8/9X9 to a uniform 9X9 core.

   Fifteen case studies were run by the Fuel Vendor to validate the stability of the Cycle 8 core. This was accomplished utilizing one dimensional and three dimensional modeling codes. The calculations were performed under a variety of conditions including but not limited to: August 15, 1992 restart conditions, August 2, 1992 restart conditions before and after FCV closure, restart conditions with worst case under corrective action restraints both now (500 MWD/MTU), at 1000 MWD/MTU, and at 1500 MWD/MTU.

   The results of these stability analysis show decay ratios for this core to be less than 1.00. All cases showed decay ratios to be between 0.2 and 0.6 indicating all situations to be self dampening.

2. In an effort to minimize the inlet sub-cooling, which can contribute to power oscillations, a change was made to the Minimum Feedwater Temperature Curve in Plant Procedure, PPM 3.1.2, "Reactor Plant Cold Startup."

3. Procedures for monitoring power oscillations will require that the ANNA system be operable and in service from greater than 25% reactor power and less than 50% core flow (see attached Figure 2).

4. In addition to general precautions recommended by the BWROG on stability, specific requirements were put in place to minimize plant testing and time spent below 50% core rated flow and above 25% core rated power. This will require shifting recirculation pumps from 15Hz to 60Hz speed at power less than 33%.

5. An approved startup plan was written and approved that controlled and specified rod patterns for the Cycle 8 startup. This plan utilized a rod pattern that was analyzed for stability prior to closing the first flow control valve for recirculation pump shift to fast speed. Revisions, except for minor rod position deviations due to instrument or equipment malfunctions, will require a new stability evaluation of the alternate rod pattern and Plant Operating Committee approval prior to recirculation pump upshift.

6. These actions were implemented by procedural changes and training of personnel on these changes. The Power to Flow Map was revised to reflect these changes by designating an "INCREASED AWARENESS" region (see attached Figure 2). BWROG Implementation Guidance for Stability Interim Corrective Actions have been implemented in Plant procedures and reinforced by training sessions and exam testing.

7. Plant Shutdown and Abnormal Condition Procedures were modified to provide increased monitoring, precautions, and direction regarding potential core instabilities.

8. The Supply System will continue its involvement in the BWROG activities involving stability. A Supply System Principal Engineer has been spending approximately half time participating in these activities. This includes work as a primary representative on the Stability Committee ATWS/Stability Task Force and the committee involved with the long term hardware proposal.

9. A memo was issued by the Plant Manager to all Plant Personnel informing them of the significance of this event. The memo provided information on the seriousness of the event , a summary of the causes, and an outline of corrective actions.

10. Plant Procedure PPM 7.4.2.1, Power Distribution Limits has been modified to require a "T" factor adjustment prior to exceeding 25 percent core power.

11. The six APRM and eighteen LPRM input signals to ANNA have been modified to eliminate the 0.3 Hertz low pass filter. All ANNA signals now have a 5.0 Hertz low pass filter.

12. A new Plant Procedure PPM 2.1.8, ANNA Stability Monitoring System, was written to describe the operation of ANNA outside of Region C on the power to flow map.

13. A peer review by the BWR Owners Group was performed of the current WNP-2 operating practices related to prevention, detection, and suppression of power oscillations.

The Supply System will continue to evaluate the appropriateness of the operating strategies implemented as short-term corrective actions. The actions taken will be considered during the evaluations that will be made during the implementation of the long-term actions. These results may identify changes needed in the operating strategies initially established.

The following long-term design process corrective actions have been identified. These long-term corrective actions address the design process and the reviews by Supply System personnel to validate fuel vendor's calculations to ensure unstable regions are avoided throughout core operating cycles. Additionally, verification of the fuel vendor's analysis to support design reviews for future cycles will be performed.

An important aspect of the development of the long-term corrective actions and the assignment of priority to each activity was in the determination of the implications on the design process and the operating strategies. Although the WNP-2 Cycle 8 core met the required reload design criteria, the power oscillation occurred outside of the currently identified region of instability. It has become apparent that a core reload design could satisfy all of the regulatory requirements and still allow undesirable operational situations. The Supply System has concluded that, as a long-term corrective action, it will be necessary to supplement our review of the design process. Because the root cause identified problems with operating strategies, the design process and the review of those designs must address those strategies. Additionally, the Supply System recognizes the importance of continued involvement with the industry in the resolution of the core stability issues. The following is a summary of the changes to be evaluated in order to supplement the existing design process.

1. The scope of the Supply System design reviews will be expanded to provide additional oversight of vendor reload design and analysis. This expanded review process will be implemented for the Cycle 9 reload design and will include increased scope and technical depth of the design. A plan for the design review for Cycle 9 will be developed by March 15, 1993 and the enhanced reviews will include:

   a. increased emphasis on the operating performance of the core in addition to meeting the licensing requirements;

b. increased awareness of the impact of core and fuel design changes on plant operations.

c. increased attention to core stability and thermal hydraulic characteristics.

d. fuel vendor will be required to perform additional stability analysis beyond the current licensing requirements.

2. The Supply System will evaluate the feasibility of changing the fuel design to one that is more stable than the current 9x9-9X design. The long-term objective is the use of fuel designs which create known and manageable stability characteristics during plant operations and transients. The goal is that the next fuel fabrication contract, for fuel to be delivered in 1995, will meet the criteria necessary to satisfy this objective. The potential of earlier changes in fuel design will be evaluated, but to assure understanding of the impacts of a new fuel design on the existing mixed core, the Supply System does not expect to be able to implement fuel design changes prior to the 1995 fuel delivery. This evaluation will be completed by March 15, 1993.

3. To support the enhanced reload design reviews and implementation of operating strategies, the Supply System will pursue obtaining core stability analysis codes. The Supply System will evaluate the existing codes and their availability in an attempt to implement their use in support of Cycle 10 core design review. This evaluation will be completed by January 1, 1994.

4. The Supply System will encourage the fuel vendor to accelerate the validation of the present stability code used for assessing selected rod patterns for the startup plan.

5. The stability of the existing core will be evaluated as part of the startup plan discussed above under short term corrective action (paragraph B.5) to ensure that the corrective actions are valid during plant startups that may occur for the remainder of Cycle 8 operation.

Beyond those issues directly related to core design and operating strategies, the Supply System identified some contributing factors that require further corrective action.

6. The frequency specified for the surveillance requirement for power distribution limits and the determination of the "T" factor for APRM set points is condition-based and leads to some confusion. The Supply System will pursue a Technical Specification change to eliminate the confusion and to eliminate the requirement for calculating "T". A Technical Specification change request will be submitted by January 31, 1994 following completion of the necessary analysis.

7. A short-term corrective action involved splitting signals for the LPRMs input into ANNA, the stability monitoring program. This implementation approach associated with the modification decreased the flexibility of ANNA. The Supply System will evaluate the impact of this reduction in flexibility. This evaluation will be completed by January 1, 1993.

8. A reliability improvement evaluation will be performed of the core stability monitor (ANNA). This investigation will include assessment of power supply, CPU redundancy, auto alarm features, and enhanced surveillance techniques to verify continued hardware and software operability. This evaluation will be completed by January 1, 1993.

9. The Reactor Engineering Group within the Plant Technical Department provides on-shift direction to the operating crews during power operation and maneuvering. The Fuels Engineering Group, within Engineering, is the primary interface with the fuel vendor. The Supply System will evaluate this division of responsibilities and the working relationship between the two groups in establishing strategies during start up and full power operation. This evaluation will be completed by March 1, 1993.

10. Replace Flow Control Valves (FCVs) and two speed pump operation with Adjustable Speed Drive (ASD) pumps. This will eliminate the need to conduct operations under the restrictions on FCVs and 15 Hz speed pumps. The current two speed recirculation pumps will be powered from adjustable speed power supplies, allowing continuous flow adjustments from 15 to 60 Hz. With this modification, recirculation flow control valves are not required. Due to implementation restraints and concerns, this modification is now scheduled to be complete by June 30, 1994. To regain flow margin for cycle 9 the Supply System is aggressively pursuing jet pump cleaning for the next refueling outage.

11. An evaluation will be performed of long term shutdown strategies to ensure the correct procedures are in place for all conditions. This action will be complete by January 1, 1993.

12. For cycle 9 a revalidation of the startup plan will be performed to assure this approach provides adequate margins for stability. This will be completed January 1, 1993.

13. A review will be performed to identify actions to be taken to improve the effectiveness of the Supply System's participation in industry activities. This will be completed by February 1, 1993.

14. Management issues associated with this event are being addressed and corrective actions are ongoing. The Management Oversight and Risk Tree (MORT) analysis has been completed.. Management has reviewed this report and initiated the following actions:

    a. Action is being taken to strengthen programs and practices used to review and assimilate industry information. Specific changes to plant practices have been incorporated to ensure BWROG and NUMARC information is critically screened for specific and for general relevance to WNP-2. An examination will be performed to identify the need to review other documents which may strengthen the influence of industry information on Supply System practices. This will be completed by December 1, 1992.

b. Action will be taken to strengthen our reactivity management program. Our existing process will be reviewed and contact will be made with other utilities to emulate the best features of their programs. This will include a review of the clarity of responsibilities between the Supply System and the contractor. In addition, the Supply System recognizes that lessons learned in this event ware applicable to other areas of our operation. Programs exist to ensure design constraints are integrated into operating practices. An assessment will be conducted to ensure the objective of a strong link between design bases and operating constraints is met. These actions will be completed by April 15, 1993.

c. A corporate level review be made of the overall relationship between Engineering Services, the fuel vendor, operators, STAs, and the SNEs to assure that responsibilities and duties for all aspects of fuel design, operation, fuel design related independent review, quality assurance, limit setting and recommendations on operating modes is well defined, active, effective and understood by all concerned. Included in this evaluation will be a review of all barriers that are assumed to be in place (such as ANNA) to prevent reactivity related events are actually being used in a fashion that the barrier is effective in performing its intended function. This item is complimentary to item 14b. This action will be complete by April 15, 1993.

15. Disciplinary action has been defined for responsible individuals at all levels of management associated with this event (see reference 6).

<u>Safety Significance</u>

This event has safety significance since 10CFR50 Appendix A, General Design Criteria for Nuclear Power Plants was challenged. Criteria 12, Suppression of Reactor Power Oscillations, states, "The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliable and readily detected and suppressed." Although exact wording of the criteria were met at all times the fact that the oscillations occurred unexpectedly results in some safety significance.

A water chemistry analysis shortly after the event showed no evidence of fuel failure. Analysis of the primary coolant gave no indication of iodine spiking which would be indicative of fuel failure following depressurization. A comparison of the coolant chemistry with previous shutdowns showed nothing abnormal.

During the initial assessment following the event, a bounding POWERPLEX analysis was performed to ensure minimum critical power ratio limits had not been exceeded. The available APRM data indicated the maximum neutron flux magnitude was approximately plus to minus 9 percent. As a bounding case this steady state analysis assumed power was increased by 9 percent with no increase in flow. The results showed a delta CPR of 0.5. With an initial CPR of 1.946, the minimum CPR would be below the Operating Limit Minimum CPR of 1.795 but well above the Safety Limit Minimum CPR of 1.07 including uncertainties in data and assumptions.

Transient analysis was also performed based on the peak to peak oscillations noted on the Local Power Range Monitors (LPRMs). This transient analysis was performed with the Supply System VIPRE Transient code and was independently performed by Siemens Power Corporation using the XCOBRA-T code. A conservative hot channel analysis using a 30 percent peak-to-peak input resulted in a delta CPR of approximately 0.20 from both codes.

It can, therefore, be concluded the oscillations did not result in any fuel failures or Safety Limit Minimum Critical Power Ratio (SLMCPR) limits being exceeded.

The safety significance of the "T" factor adjustment not being performed prior to exceeded 25 percent power was evaluated. The purpose of this adjustment at non rated (low power) conditions is to provide added protection against a highly peaked power distribution by temporarily adjusting the APRM sensitivity to a more conservative value. Had this been done at the 25 percent power point it would have resulted in a 15 percent decrease in the margin to the trip setpoint. This is not safety significant.

The safety significance of the 0.3 Hertz filters associated with the ANNA input has been reviewed. If ANNA had been used with these filters it would have provided results that were non-conservative. The ANNA monitoring system was one of the systems being relied upon for the "detect and suppress" strategy associated with core oscillations when operating in Region "C" on the power to flow map. The inoperability of this equipment because of filtered input was safety significant since it could have allowed the plant to go into the region of instability unknowingly if the system had been used. However, it is backed up by the reactor protection system (high neutron flux and flow referenced flux scram) which would have shutdown the reactor if the oscillations became too severe.

<u>Similar Events</u>

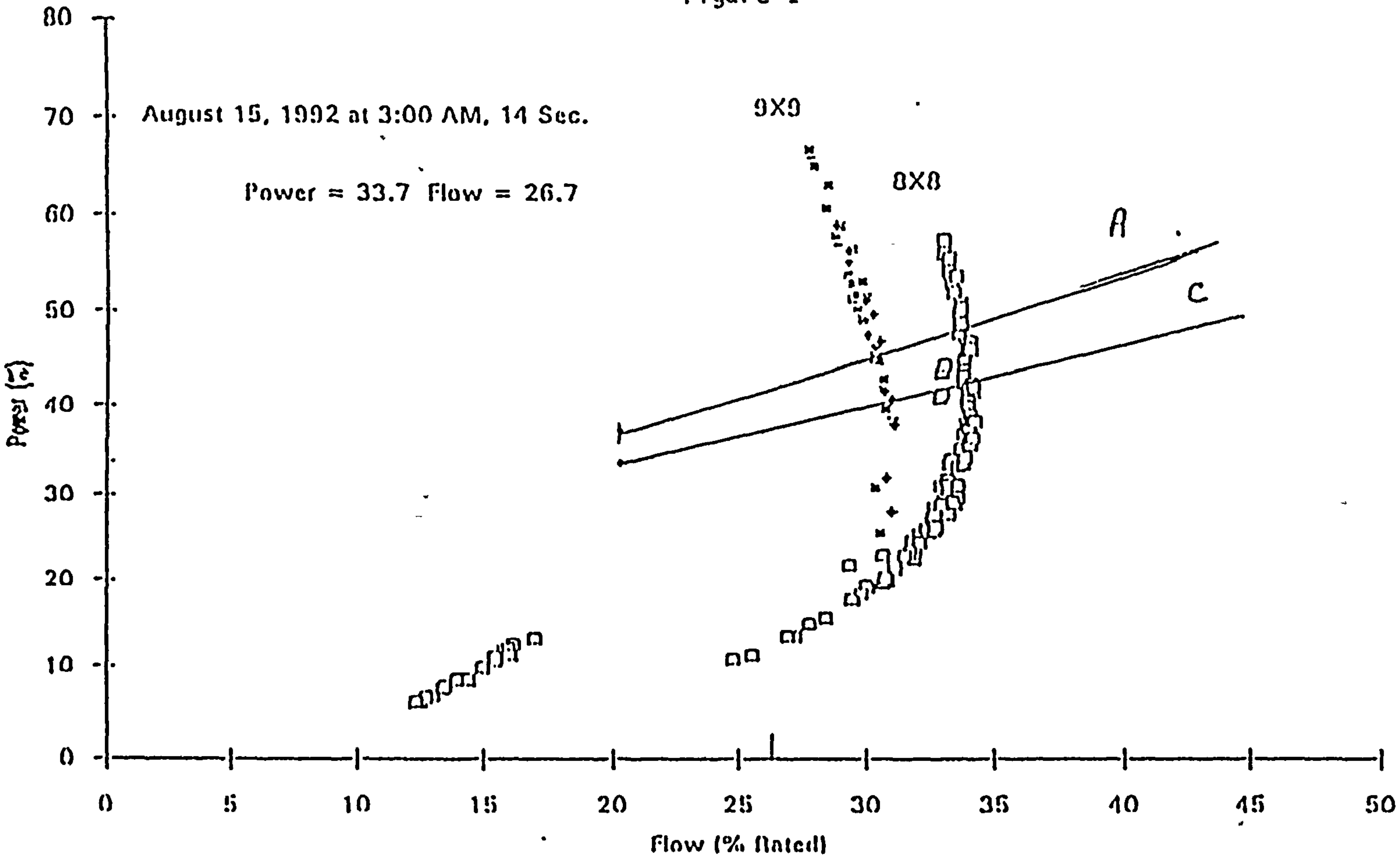
There have been no similar events at WNP-2.

EIIS Information

Text Reference	EIIS Reference System	EIIS Reference Component
Average Power Range Monitors (APRMs)	IG	MON
Local Power Range Monitors (LPRMs)	IG	MON
Recirculation Flow Control Valve	AD	FCV
Reactor Bottom Head Drain Valve Bypass Valve (RWCU-V-103)	CE	V
Advanced Nuclear Noise Analysis (ANNA) Monitoring System	JC	--

Power Flow - Individual Assemblies

Figure 1



LICENSEE EVENT REPORT (LER) TEXT CONTINUATION											
FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (8)			PAGE (3)		
		Year	Number	Rev. No.			
Washington Nuclear Plant - Unit 2	0 5 0 0 0 3 9 7	9 2	0 3 7	0 1	17	OF	17
TITLE (4) MANUAL REACTOR SCRAM DUE TO CORE INSTABILITY							

Two Loop Operation
Plant 2 Operating Map

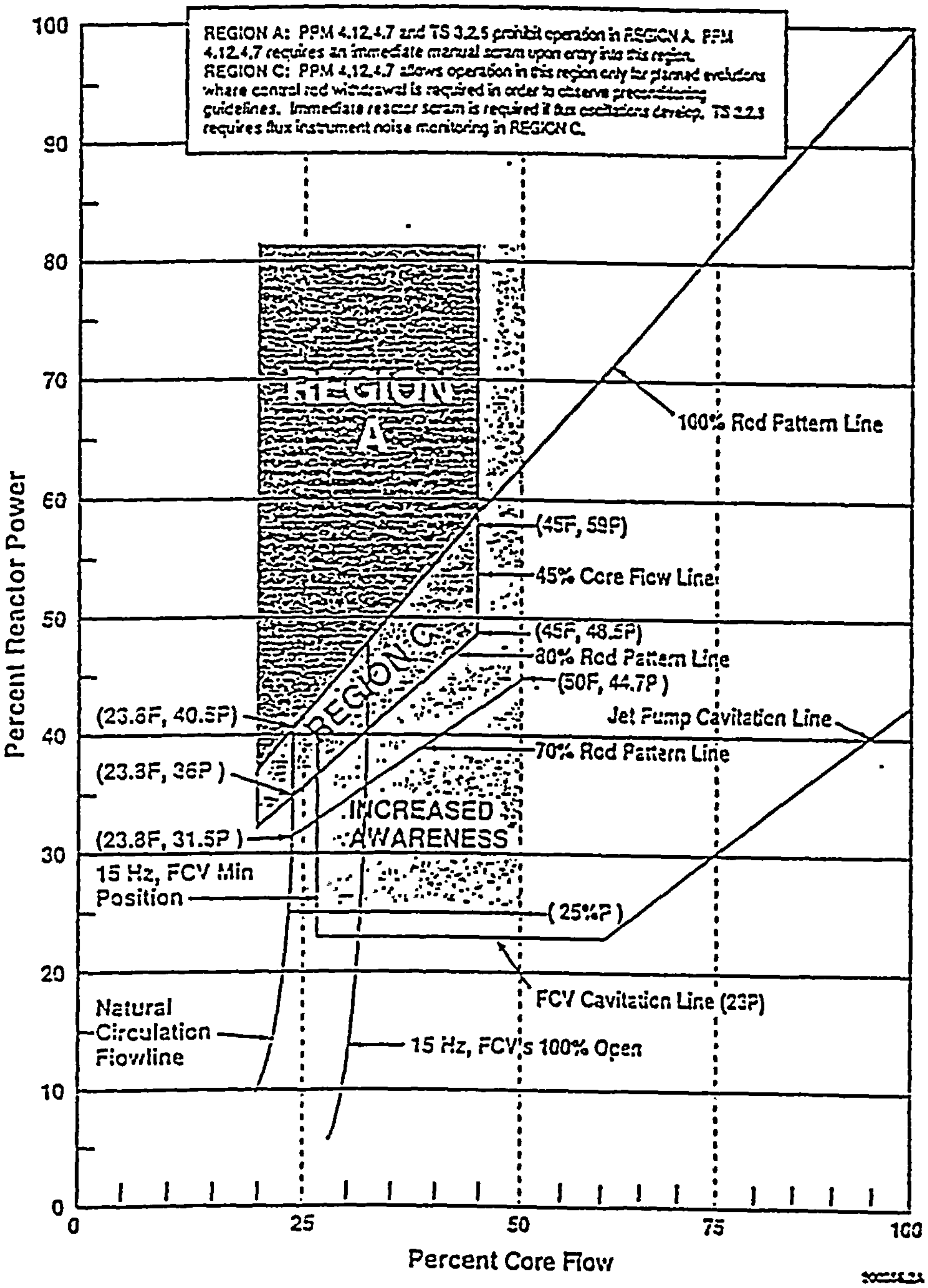


Figure 2