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MARTIN,J.B. Region 5 (Post 820201)

SUBJECT: Responds to violations noted in Augmented Insp Team Rept 50-397/92-30.Corrective actions:failure to perform technical work at acceptable level of detail & quality addressed.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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October 30, 1992
GO2-92-247

Docket No. 50-397

Mr. J.B. Martin,
Regional Administrator
Region V
U.S. Nuclear Regulatory Commission
1450 Maria Lane
Walnut Creek, Ca. 94596

Dear Mr. Martin:

Subject: **NUCLEAR PLANT NO. 2, OPERATING LICENSE NPF-21 RESPONSE TO NRC AUGMENTED INSPECTION TEAM REPORT**

Reference: Letter dated September 29, 1992, JB Martin to AL Oxsen, "NRC Augmented Inspection of Washington Nuclear Project, Unit 2"

My staff and I have reviewed in detail the U.S. Nuclear Regulatory Commission (NRC) Augmented Inspection Team (AIT) report regarding the core power oscillation event which occurred at WNP-2 on August 15, 1992.

We recognize that this event was avoidable, and as Supply System management accept responsibility for not avoiding it. Coming on the heels of the Requalification Program failure, this event has made us keenly aware of the need to truly approach the conduct of our licensed activity in a proactive and self-critical manner. To that end, we have increased benchmarking to align objectives of our programs, heightened sensitivity to the need for management accountability, conducted self-critical appraisals of our program/functional areas, and redirected our programs to identify design bases constraints. Overall, we have taken steps to responsibly address the weaknesses identified and to become more proactive at avoiding events in the future.

Your cover letter, which transmitted the AIT report, cited fundamental errors which contributed to the August 15 core power oscillation event. In the October 15 Management Meeting with

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you and your staff, we acknowledged we should have responded to prior industry information and critically questioned our design oversight and operating philosophy to minimize the potential for core power oscillations.

We recognize management emphasis on accountability and accomplishing objectives needs improvement. The actions we had initiated to move in that direction did not yield satisfactory results in a timely manner. In that light, we have decided to take aggressive personnel actions intended to demonstrate to the entire organization that failure to perform carries significant consequences and that we are serious in our efforts to improve accountability within the Supply System. We intend that this signals a change in direction for the Supply System. We are now forcefully dealing with issues of personal responsibility by dealing with individuals who do not meet our expectations. We have identified the responsible individuals, at all levels of management, who should have provided the management oversight and created barriers to prevent the conditions which led to the event and are taking significant disciplinary actions to reenforce the seriousness of the shortcomings in management effectiveness. These actions include counseling sessions, letters of reprimand, and forfeiture of compensation in varying degrees depending on level of responsibility.

Our analyses of the contributing factors have quantified the relative contributions of each factor and the magnitude of these factors has guided the development of our corrective actions. In summary, the impact of the extreme power distribution established during the August 15 startup consumed approximately 75% of the available stability margin while the core design (i.e. single rod sequence loading) contributed about 10-15% reduction and the bundle design (9x9) a complementary 10-15% loss. As a consequence, the main focus of our efforts has been directed at the sufficiency of power distribution controls in the short term. In addition, we are examining the design process implications of this event to deal with the two lessor contributors in the long term.

The failure to establish adequate operational constraints to counter-balance the reduced stability margin in our core/fuel design resulted from the responsible individuals not properly comprehending the implications of our core design (both the 9x9 fuel and the single rod sequence core loading) and its associated sensitivity to extreme startup power distributions. We recognize this as a serious breakdown in the fundamental responsibility of our design organization. We have addressed this shortcoming in the fuel design organization by strengthening the design review process and becoming involved with our fuel vendor's design activities. In addition, we have introduced new requirements for the fuel vendor to perform

much more in-depth stability analyses on their cycle specific core designs to ensure we understand the limitations/sensitivities of our designs. We will also modify, as appropriate, the Core Operating Limits Report (COLR) to specifically detail the stability related operating constraints for the Plant Nuclear Engineering Organization.

For future reload fuel supply we will utilize, as an important decision criterion, relative stability performance of proposed designs. State-of-the-art stability methods will be employed in these evaluations.

Our on-shift Station Nuclear Engineer (SNE) exercised poor judgement in establishing the highly skewed power distribution that contributed to the onset of core power oscillations. Given the experience and knowledge of our Reactor Engineering Organization, another independent opportunity to bring a sensitivity to the potential for core power oscillations from power distribution effects was missed in that they failed to seek fuel design limitations from the Engineering Organization. As a result, our operations staff was not provided with clear operating parameters which would minimize the potential for core power oscillations. Similarly, our Shift Technical Advisors (STA's) and SNE's were not provided with explicit proceduralized operational bounds which could be used by our Senior Reactor Operators to assess STA/SNE performance.

Our evaluation of this event, in conjunction with the input we obtained from the Industry Peer Review of our corrective actions, indicates that we need to provide better guidance to our staff on reactivity management. We have initiated actions to review our existing guidance, have initiated contact with other utilities to emulate the best features of their programs, and have revised the expectations and relationships of those organizations involved with nuclear safety of the core.

Our shortcomings in analyzing industry information from the BWROG Stability Committee and ensuring that adequate operational precautions were taken highlight a process weakness which is being addressed through appropriate process changes. More fundamentally however, this experience identified that our involvement in industry activities is not providing, generically, the course corrections/input into our programs and practices from lessons-learned that it should. To address this organizational performance weakness, we will perform a critical examination of the shortcomings in this area and institute mechanisms to strengthen the influence of industry initiatives on our practices. We have begun a series of benchmarking activities for our major programs. This effort will continue with renewed emphasis on creation of change and alignment

of our objectives with those of other successful programs. In addition, we have been conducting both functional and program reviews to measure progress and to confirm alignment of objectives with expectations. These reviews have been and will continue to be conducted in both Operations and Engineering.

We share your concern regarding our continued weakness in responding to generic industry concerns, with this event and operator requalification being examples. As we discussed in the October 15 management meeting, it is easy to rationalize--as we did in the case of our licensed operator training program--adequate performance based on a review of a generically worded industry notice. As we stated in the meeting, our logic at that time was "we have just gotten a clean bill of health six months ago, so this must be a problem for someone else."

Our sensitivity to generic information must improve. We must ensure that our ongoing processes identify safety significant information and that we then provide a rigorous evaluation for applicability to WNP-2. This review must lead to implementation of effective training and procedure revisions where appropriate.

It is clear that the way to avoid these issues is to stay in touch with industry events and to benchmark our performance against the best performers. We have not done as well as we must in this area. However, as we pointed out in our October 15 meeting, we are currently meeting more with industry groups and industry peers to learn from their experiences. We believe the expanded effort to visit and learn, coupled with our increased sensitivity to ask "why can't this event (or problem) happen here," will greatly reduce the potential for future issues where we fail to heed generic information on industry issues.

On a more generic basis, we recognize that we must apply the lessons learned from the core power oscillation event to other areas of our operation. The question of adequate integration of design constraints into Operations will continue to be addressed through existing, on-going programs which have as principle objectives the review of design bases and proper implementation of limitations. These programs include, but are not limited to, such activities as the Design Requirements Documentation (DRD) Program, the Engineering Calculation Upgrade Program, the Engineering Setpoint Program, and the Plant Procedure Upgrade Program. A self-assessment and an independent assessment of these programs will be conducted to ensure that the design bases link to operational constraint objectives of these efforts is being met.

In conclusion, I reiterate that we recognize past shortcomings in the areas addressed in your letter and are taking prompt action to deal with those issues. While addressing the specific issues associated with the core power oscillation event, we are looking beyond the specifics of this event to apply the lessons learned to prevent similar problems elsewhere.

All this having been said, I wish to reemphasize what I believe is the most important step--the personnel actions--we are taking in the wake of the core power oscillation event. We intend this to be a sea change for the Supply System, a strong signal that Supply System management has high expectations of its nuclear team and will hold that team accountable for performance shortcomings.

We have attached, for your information, comments related to the technical details of the AIT Report.

Very truly yours,

A.L. Oxsen
Acting Managing Director

Attachment: As stated above

cc: JW Clifford, NRC
Document Control Desk, NRC
NS Reynolds, W&S
DL Williams, BPA (399)
NRC Site Inspector (901A)

ATTACHMENT 1

WNP-2 Core Instability Event of August 15, 1992

Comments on the Augmented Inspection Team (AIT) Report

Report Number 50-397/92-30, September 29, 1992

The following comments/corrective actions are respectfully submitted in response to the AIT report:

1. COVER LETTER AND SUMMARY

AIT Report Transmittal Letter, Second Page, First Paragraph

Refer to: "All of these errors in the control of the nuclear fission process appear to arise from a failure to perform technical work at an acceptable level of detail and quality."

Comment: The Supply System agrees that our performance in this important area requires improvement. The cover letter transmitting this response to the AIT report outlines several actions the Supply System is taking to address this issue. These include improved core design oversight, improved industry involvement and consideration of generic industry advisories, as well as the development of an integrated corporate-wide Reactivity Management Program to strengthen controls on core management activities.

Page 2, Sixth Bullet (also pages 8-11)

Refer to: "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."

Comment: The Supply System acknowledges that the BWROG advisory letter did not receive a complete evaluation for potential impact on WNP-2 operation. While there may be some question as to the clarity of advise being given in the BWROG letter, the Supply System recognizes that had we been more involved with the stability committee and their discussions related to the potential for instabilities outside of Region C, we would have taken steps to incorporate appropriate guidance in our procedures.

ATTACHMENT 1
(Continued)

Page 3, Top Bullet

Refer to: "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."

Comment: Engineering, and in this case more specifically Nuclear Engineering, has overall responsibility for the core design and for establishing operational limits required to remain within the design and licensing bases. As correctly pointed out in the AIT Report the Nuclear Engineering organization failed to provide limits or guidance for plant procedures to ensure that operation of the plant would remain within the assumptions used in the design bases. To correct this deficiency, the Nuclear Engineering department will incorporate into its core design review plan a requirement to identify and understand the core design criteria or safety analysis assumptions used. They would transmit applicable limitations, criteria, and recommendations to the plant for incorporation into plant procedures. These limitations will be either incorporated into the cycle specific COLR by Engineering or transmitted to the plant through the Engineering Transmittal commitment process. (Please see LER 92-037-01, Corrective Action Number 9)

2. DESCRIPTION AND ANALYSIS OF THE EVENT

Page 6, third paragraph of 3.1.3

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

Comment: The SNP bounding analysis was performed with the XCOBRA code to support the Supply System's efforts to evaluate the oscillations effect on fuel integrity. This analysis resulted in a CPR change of 0.37 and an MCPR of 1.58 as reported to the AIT on August 27, 1992. As concluded by the AIT, this was well above the SLMCPR of 1.07, but exceeded the Operating Limit minimum CPR of 1.795. If this bounding analysis is the basis for the Minimum Critical Power Ratio (MCPR) of 1.68 reported on page 4, paragraph 4, the correct MCPR should be 1.58.

ATTACHMENT 1
(Continued)

3. REVIEW OF GENERIC CORRESPONDENCE

Page 11 second bullet

Refer to: "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."

Comment: The filtering mentioned in the BWROG letter dated Nov 3, 1988 and NCRB 88-07 Supplement 1 refers to the filtered APRM signals that provide the Thermal Power Monitor reference signal for the flow biased scram circuit(approximately 6 second time constant). These filters render the automatic scram protection ineffective in suppressing regional instabilities. The Supply System recognized the ineffectiveness of the automatic scram protection during regional oscillations and complied with the bulletin by requiring a manual scram on a two pump trip event or during any event that placed the plant in Region A. Moreover, measures were taken to provide operator training and procedural guidance for instability recognition and suppression.

The stability monitor, ANNA, is not affected by the Thermal Power Monitor filters, but by the filtering in the signal conditioning that exists at the front end of the Process Computer(0.3 Hz).

Page 11, Second Bullet

Refer to: "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."

Comment: The identification of the ANNA filter concern occurred on Saturday August 16, 1992 during our investigation and confirmation of data activities. We had not completed our assessment of the impact of this situation by the time the AIT team presentation was provided on Monday, August 18. We stated very clearly in our presentation that our results were preliminary and further evaluation was to occur following completion of our data integrity efforts. A statement in paragraph 5.4.1 of the AIT report confirms that the filter issue was identified in parallel with the AIT investigation.

Page 11, Forth Bullet

Refer to: "The AIT found that the licensee's program for control of generic information had broken down: it permitted the procedures and training to conflict."

Comment: BWROG interim guidance was received by WNP-2 between 18 March and 28 April 1992 time frame. It was reviewed by the Stability Committee member (both Analytical and Hardware Groups), and the Reactivity Controls Committee member. It was reviewed at the supervisory level for its applicability to WNP-2 at this time. A management decision was made that this information should be disseminated to STAs, Operators and SNEs. It was from this decision that a training module was created and implemented (82-SQT-9202-L2). Likewise, a conscious decision was made that this information was not required to be implemented into plant policy and procedures due to its interim nature. The program for controlling generic information did not break down as the Owner's Group correspondence does not normally go into this system as would SOERs, INs, or SENs. These items would receive a more formal review for applicability and implementation. In hindsight, the decision to implement the BWROG interim guidance in training only and not in plant policy and procedures was flawed. An evaluation of this process is being conducted by Technical Assessment to determine the merits of including Owner's Group guidance into the OER review process.

4. REVIEW OF OPERATOR AND STA PERFORMANCE

Page 13, Second Bullet

Refer to: "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."

Comment: PPM 9.3.9, Control Rod Sequence Development and Control, requires calculated power distributions for the "target rod pattern" only. This refers to the full power target rod pattern which will be used for the next exposure increment following

the rod set evolution. The STA/SNE was not required by procedure to have calculated power distributions for intermediate patterns. However, the procedure was clearly inadequate.

Page 15 second paragraph

Concern: "The data on 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."

Comment: Cases 7 and 8 of Table 1 represent "Offline" calculations that were performed after the event, to reconstruct the existing power distributions. Cases 7 and 8 were not available to the crew for monitoring. However, the Supply System admits that they were not sufficiently sensitive to changes in power distribution.

Page 15, Third and Forth paragraph

Refer to: "The 'Power Distribution Constraints' section of procedure PPM 9.3.12 stated that:....A (Core Maximum Peaking Factor) of approximately 3.4 usually results in the optimum rod pattern... The crew did not attempt to limit CMPF to 3.4..."

Comment: PPM 9.3.12, Reactor Power Maneuvering, provides an example of a peaking factor which has historically proven to provide a bottom peaked power distribution and yet provide adequate margins to PCIOMR limits following the recirculation pump shift from 15Hz to 60Hz with drive flow increased to 24,000 GPM/loop (to minimize pump vibration). This peaking factor (i.e. 3.4) is not stated as, nor intended as a limit. The STA/SNE strategy in rod pattern selection was consistent with the technical basis behind the guidance provided in PPM 9.3.12 so there was no violation of the procedure. However, the Supply System agrees that the procedure was inadequate.

Page 17, First Full Paragraph

Refer to: "The inspectors found that there were no administrative controls requiring peer review of SNE changes to prescribed control rod order sheets."

Comment: Changes in our operation are being made to require independent review and verification by our Fuel Engineering group of control rod sequences developed or modified by the shift nuclear engineers. The purpose of this review is to ensure that the patterns comply with the core design bases. Plant procedures PPM

ATTACHMENT 1
(Continued)

9.3.9 "Control Rod Withdrawal Sequence Development and Control" and PPM 9.3.10 "Control Rod Sequence Exchange" which address target control rod patterns will be revised to incorporate this change. In addition PPM 9.3.12 "Plant Power Maneuvering" which governs deviations from these patterns will be revised to require additional independent reviews if the deviations fall outside predefined allowable limits. Corrective action is being taken to upgrade the reactivity management process as described in LER 92-037-01, Long Term Corrective Action Number 14b.

5. ASSESSMENT OF ENGINEERING PERFORMANCE

Page 20, Table 2

Refer to: "Two pump minimum flow intercept with 100% rod pattern line-Region A exclusion boundary."

Comment: Table 2 indicates that statepoint 47%P/27.6%F corresponds to the "two pump minimum flow intercept with 100% rod pattern line-Region A exclusion boundary". Statepoint 47%P/27.6%F is above the 100% rod line and is within region A.

Page 20, Table 2

Refer to: "natural circulation flow intercept with 100% rod pattern line-Region A exclusion boundary."

Comment: Table 2 indicates that statepoint 42%P/23.8%F corresponds to the "natural circulation flow intercept with 100% rod pattern line-Region A exclusion boundary". Statepoint 42%P/23.8%F corresponds to the natural circulation flow intercept with the APRM rod block line. This is above the Region A lower exclusion boundary.

Page 20, Second and Third Paragraphs

Refer to: "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."

Comment: The NRC issued the Safety evaluation of the SNP stability methodology on May 10, 1984. The Reference document describing the methodology is XN-NF-691(P) and Supplement 1. The conclusions of the safety evaluation are:

"The staff approves the Exxon stability methodology for use in licensing reload fuel under either of the following conditions:

1. The calculated decay ratio for the proposed cycle is less than or equal to 0.75 and acceptable Technical Specification restrictions are placed on natural circulation operation; or;

2. The calculated decay ratio for the proposed cycle is less than or equal to 0.90 and acceptable Technical Specification requirements are placed on natural circulation and single loop operation including proper surveillance of both LPRMs and APRMs."

This COTRAN SER defines regions where operation is not allowed (power/flow regions where DR is greater than 0.9 and natural circulation operation and single loop operation when DR is greater than 0.75). The SER also identifies regions where operation is allowed when detect and suppress operation is being performed (power/flow regions where DR is greater than 0.75 and less than 0.9). Natural circulation operation is not allowed at any time.

The COTRAN SER together with NRC Generic Letter No. 86-02 and NRC Bulletin No. 88-07, Supplement 1 were used to establish criteria for defining acceptable operating regions for WNP-2. The criteria defining operating regions A, B, and C at WNP-2 are stated in the WNP-2 licensing response to IEB 88-07, Supplement 1; G02-89-030, Response to IE Bulletin 88-07, Supplement 1, Power Oscillation in Boiling Water Reactors, March 3, 1989; G02-89-051, Support of the Supply System Response to IEB 88-07, Supplement 1, March 31, 1989; G02-89-101, Request for Amendment to Technical Specifications 3/4.2.6, Power Flow Instability and 3/4.2.7 Neutron Flux Noise Monitoring, Supplemental Information, June 2, 1989, and the NRC Safety Evaluation Report, SER, Issuance

of Amendment No. 71 (TAC No. 72924), June 23, 1989. These criteria which are consistent with the recommendations of General Electric Service Information Letter (SIL) No. 380, were reviewed in 1989 and subsequently accepted when the ANNA stability monitor was approved for use at WNP-2. These criteria

potentially result in operating regions that are more restrictive than the regions defined in the GE SIL 380 notification and the results reported in Table 2 confirmed that the Region A-C boundary remained applicable for Cycles 7 and 8.

Quoting from the WNP-2 Licensing Submittal:

"...Those portions of Region B which fall outside a region of 0.9 decay ratio as calculated by Advanced Nuclear Fuels (ANF) (in support of reload license applications) will be redefined as part of Region C, where operation is allowed under a controlled manner when fuel preconditioning is required. Those areas below the 100% rodline for which the calculated decay ratio is greater than 0.9 will remain defined as Region B, where planned operation is prohibited..."

Quoting from the NRC Safety Evaluation:

"2.0 EVALUATION

The IRSA specify three regions (A, B, C) on the power-flow map involving different degrees of allowed or prohibited operation. These are bounded by constant flow lines or control rod lines... Region A is above the 100 percent rod line ... and below 40 percent flow. Region B is between the 80 to 100 percent rod lines and below 40 percent flow. Region C is above the 80 percent rod line and between 40 and 45 percent flow...

Specification 3.2.6 and Figure 3.2.6-1 provide, in both the current and proposed TS, a prohibited region which is larger than the IRSA region A. In the new TS, it extends to the 45 percent flow line, thus covering the upper part of IRSA region C. It can also extend below the 100 percent rod line since the lower boundary is the more conservative of the 100 percent rod line or a line below (ANF) calculated 0.90 oscillation decay ratio (DR). The proposed TS required a manual scram exit from the region, in accord with the IRSA. These, and other wording changes of this Specification, are reasonable, generally conservative compared to the current Specification or IRSA, and are acceptable..."

SPC used the criteria defined above to identify acceptable stability regions at WNP-2. The calculated Decay Ratios identified in Table 2 of the AIT report satisfy the acceptance criteria specified and accepted above. These criteria were reviewed and accepted by the USNRC during the review and acceptance for use of the ANNA stability monitor at WNP-2 in Cycle 5.

Page 21, second paragraph, first and second sentences

Refer to: "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."

Comment: ANF believed that the 9x9 fuel had some characteristics which could improve stability and some characteristics which could reduce stability, but ANF did not conclude that ANF 9x9 fuel was less stable than existing designs of 8x8 fuel. The AIT report should also provide a brief summary of results of these tests. An independent review of the measured data by ORNL concluded that "the 9x9 fuel does not produce major changes in stability behavior compared to BWRs loaded with standard 8x8 fuel." (ORNL/NRC/LTR-91/12 dated August 1991). Without a summary of the test results, the reader is left with the incorrect impression that ANF 9x9 fuel is less stable than 8x8 fuel.

Page 21, second paragraph, last sentence

Refer to: "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."

Comment: "The histogram provided to the AIT during their inspection shows that the 9x9-9X has a two-phase/single-phase pressure drop ratio slightly higher (approximately 3.8%) than the GE8R. It was also stated that this higher hydraulic pressure drop ratio was mitigated by the less negative void coefficient and the reduced axial peaking. The LAPUR sensitivity calculations (Appendix C) only considered the hydraulics contribution to the stability. When all factors affecting stability are included, the 9x9-9X fuel would have comparable stability performance to other fuels in US BWRs.

ATTACHMENT 1
(Continued)

Page 21, third and fourth paragraphs

Refer to: "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 oscillation (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...."

Comment: The licensing results for Cycle 7 & 8 did not indicate a reduction in stability margin compared to previous cycles with 8x8 fuel.

COTRAN DR FROM LICENSING CALCULATIONS

Cycle	65%P/45%F	47%P/27.6%F	42%P/23.8%F
5	0.49	0.89	0.82
6	0.46	0.86	0.83
7	0.42	0.86	0.81
8	0.41	0.77	0.64

An instability event such as was experienced by WNP-2 is not part of normal operation. Operating procedures require reactor scram to terminate such instabilities.

Page 25, First Paragraph

Refer to: "The licensee selected SNP as both the Cycle 8 reload designer and the fuel vendor. SNP was an approved vendor on the licensee's approved suppliers list based on both Supply System audits and industry audits."

Comment: The AIT report implies that the Supply System makes an annual decision regarding which vendor will fabricate and design the WNP-2 reload core. In fact, the contract with Siemens Power Corporation was awarded in October of 1971 and covers all reload fabrication

for WNP-2 through 1994. The contract was modified to give the Supply System the option to assume design responsibilities during the last two years of the contract; 1992 and 1993. Continuing NRC review of the Supply Systems methods is underway.

Page 25, Section 5.2.2.2, fourth bullet item

Refer to: "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."

Comment: The SPC design verification is performed in accordance with the Quality Assurance Topical Report, EMF-1A, Revision 25, approved by the NRC. Section 2.1 of the Topical Report states that one of the requirements that is met is ANSI N45.2 (1977). Appendix I of the subject document lists any exceptions that are taken to the Regulatory Guides and daughter standards. No exceptions are taken to Section 6 of ANSI N45.2.11 which states, "This verification may be performed by any competent individuals or groups other than those who performed the original design, but who may be from the same organization."

Section 3.5 of the Topical Report states, "The individuals responsible for performing design verification should include persons other than those who performed the original design."

Implementing QA Procedure ANF-POO,002, Revision 21, Section 3.5.1, states, "Design verification shall be performed by competent individuals other than those who performed the original design and generally should not be the designer's supervisor."

The fact that input to an analysis was from a different analysis that had been performed by the designated reviewer was in accordance with the SPC program because this input would have been verified by a different individual.

While the reload design group consists of seven engineers and one technician, a review of the records shows that fifteen different engineers participated in the WNP-2 Cycle 8 reload licensing campaign.

Page 26, first bullet

Refer to: "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."

Comment: The SPC procedure for Calculation Notebooks, EMF-954, allows six months for the notebooks to be microfilmed for archiving. This is necessary to allow the engineer ready access for discussions with customers on the documents that resulted from the notebooks. These documents, when issued, are controlled with copies in the vault and the notebook becomes backup material.

Page 29, paragraph 5.3.6

Refer to: "As a result of the evaluation Siemens recommended, and the licensee accomplished, a change of the 0.3 HZ filters to 5 HZ."

Comment: The Supply System questioned SNP as to the design bases for the 0.3 Hz filter and its associated effects on the establishment of the Fuel Design Safety Limit (uncertainty in the neutron flux correlation to bundle/pin powers). When it was determined that SNP could not evaluate or quantify separately the filter contribution without considerable additional analyses, the Supply System concluded that a general change to the LPRM inputs for thermal limits evaluation was not prudent and devised a change that would only affect the LPRM inputs to ANNA. SNP concurred in this approach.

6. EVALUATION OF EQUIPMENT PERFORMANCE

Page 30, Paragraph 5.4.1

Refer to: "The assessment was flawed in that it did not consider that all but two of the LPRMs data recordings had been attenuated by about 75%..."

Comment: On August 15, between 1000 and 1300 hours, the Supply System assembled a team to begin an investigation and analyze available data. A review of the APRM data disclosed discrepancies in the magnitude of the oscillations between the PPCRS and TDAS data. This also raised questions about the adequacy of the LPRM data. While the cause of these differences was being investigated, preliminary bounding calculations using existing data were being performed. In the preliminary presentation to the AIT we indicated that conclusions were tenuous and that further data verification and evaluation was needed. The AIT report confirmed that the filter issue was being considered by the Supply System and that our final evaluation reached the same conclusions with respect to safety significance.

7. EVALUATION OF LICENSEE EVENT INVESTIGATION

Page 31, First Paragraph

Refer to: "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:"

Comment: The Root Cause Summary which was attached to the Supply System response to the Confirmatory Action Letter (CAL) indicated that a MORT analysis would be conducted to identify any management issues for the power oscillation event. Likewise, LER 92-037, stated in the cover letter that a "MORT analysis underway will identify root causes and corrective actions for the...management issues... and be reported in LER supplement by November 2." Due to the response time requirements of the CAL, the Root Cause Summary did not include these developing Management issues. These issues are included in the final Root Cause for this event to be issued with the LER 92-037-01.

ATTACHMENT 1
(Continued)

8. APPENDICES

Appendix A, Note 2

This paragraph refers to calculations performed with the VIPRE and XCOBRA-T codes. The second sentence in this paragraph states that using the measured pressure drop as a boundary condition resulted in a peak-to-peak flow oscillation of 26%. In the SPC XCOBRA-T calculation, using the measured pressure drop as a boundary condition resulted in a peak-to-peak flow oscillation of 23%.

Appendix E, Stability Calculations for the August 2, 1992 Startup

The 0.78, 1.01 and 0.68 decay ratio results are from the 3-D time domain code instead of STAIF.

Appendix F, Review of Previous Cycle 8 Startup Reactivity Anomalies

The July 4, 1992 startup (initial Cycle 8 critical) and the subsequent startup on July 10, 1992 both achieved criticality on Rod 26-07. This is a difference of 0.43% delta k/k(4.3 mk) from the pre-startup prediction.

The title of Figure F.2 states that the data shows a 10% delta k/k bounding error band. However, the data shows a 1% delta k/k bounding error band.

Appendix F Adequacy of the PER on Reactivity Anamoly

The criticality observed on July 4, 1992 occurred well within the expected tolerance for an estimated critical position. The statement is not a fair representation of the corrective actions contained in the PER. The rod sequence used complies with accepted practices and is consistent with banked rod withdrawal and reduced notch worth procedures. This reflects an appreciation for responsible operation with a control cell core and single rod sequence. The modifications to the startup control rod withdrawal sequence were made to allow a slow rate of positive reactivity addition in a symmetric fashion to minimize flux tilt. This change was not intended to correct for an early criticality. The Supply System does not believe the critical position to be outside of the expected range.

ATTACHMENT 2

This attachment provides the status (as of October 30, 1992) of corrective actions established as a result of the power oscillation event. Numbers in parenthesis following the status refer to corrective actions listed in the Licensee Event Report for this event (LER 92-037, Rev 1).

Short Term Corrective Actions:

1) Implement startup power distribution constraints

    -total peaking
    -CPR
    -Decay ratio

    Status: Complete, implemented in our procedures. (1)

2) Inlet Sub-cooling constraint

    Status: Complete, implemented in our procedures. (2)

3) ANNA Operability

    Status: Complete, implemented in our procedures; see Long Term Action on ANNA reliability improvements. (3)

4) Minimize Time at Lower Power

    Status: Complete, implemented in our procedures. (4)

5) Approved Startup Plan with Restricted Rod Patterns & Deviations

    Status: Complete, implemented in our procedures; see Long Term Action on Confirmation of Approach. (5)

6) Add Increased Awareness Zone on Power/Flow Map

    Status: Complete, implemented in our procedures. (6)

7) Increased Monitoring during Plant Shutdown

    Status: Complete, implemented in our procedures; see Long Term Action on Shutdown Strategy. (7)

8) Increased Plant Involvement in BWROG Stability Activities

Status: On-going with improved expectations/responsibilities. (8)

9) Memo to Staff indicating Significance of Event

Status: Complete. (9)

10) T Factor Adjustment procedure change

Status: Complete. (10)

11) ANNA System Input Filter Modifications

Status: Complete; see Long Term Action on ANNA improvements. (11)

Long Term Corrective Actions:

12) Increased Vendor Oversight for Reload Design

-Operating performance reviews with respect to stability
-Increased awareness core/fuel design changes on Plant Operations
-Overall increased awareness/sensitivity on Stability
-Additional Stability Analyses from Vendor required

Status: In-progress, SNP/SS Team to Improve Stability Reviews working on Methods Approach, Vendor Oversight Plan under development, Evaluation for Scatter Load verses continued Single Rod Sequence underway. (Long Term Corrective Actions #1)

13) Consider Fuel Design Changes to improve Core Stability Margin

Status: In-progress, Cycle 9 mechanical design changes evaluated with no change feasible, Ultra-flow spacer design under consideration for cycle 10, Enrichment/Gadolinia design parameters under review. (LTCA #2)

ATTACHMENT 2
(Continued)

14) Consider obtaining in-house Stability Analysis Capability

Status: In-progress, both a core design tool and an on-line monitor coupled with our power distribution monitor are being considered. (LTCA #3)

15) Encourage Vendors to accelerate validation of advanced Stability codes

Status: Completed, SNP will complete benchmark activities and submit for review in June 1993. (LTCA #4)

16) Continue Startup Plan with Stability Analyses for future startups

Status: Completed, our SNP requirements have been established, internal control procedure being prepared. (LTCA #5)

17) Create Improved Process for Review of Generic industry information

Status: In-progress. (LTCA #14)

18) Re-evaluate the relationships between the various groups involved in fuel design

Status: In-progress, outside peer review is being acquired. (LTCA #9)

19) Develop an ANNA periodic surveillance procedure

Status: In-progress, weekly surveillance procedure complete, software periodic surveillance procedure drafted and under review, periodic end-to-end hardware procedure development underway. (LTCA #6)

20) Prepare a Tech Spec Change on T Factor Adjustment to change approach

Status: In-progress, RFP to SNP being specified. (LTCA #6)

21) Evaluate Improvements to ANNA Useability/Reliability

Status: In-progress, software has been adapted to other computer systems, will be operable by Jan 1, 1993, Operator alarm feature will be operable by Dec 1, 1992, Software addressable LPRMs under evaluation. (LTCA #8)

ATTACHMENT 2
(Continued)

22) Conduct a Peer Review of our Nuclear Engineer Processes

Status: Completed, main recommendation to implement a revised Reactivity Management Program underway. (13)

23) Replace Recirc FCVs with ASDs

Status: In-progress, construction has begun, Licensing work underway, expect final implementation in R9. (LTCA #10)

24) Create a Corporate-wide Reactivity Management Program

Status: In-progress, target completion by 4/15/93. (LTCA #14)

25) Examine Long Term Shutdown Strategy

Status: In-progress, both operating changes as well as analytical/power distribution constraints under development. (LTCA #11)

26) Confirm Startup Plan and Limits Approach Remains Applicable

Status: In-progress, considerable work/coordination with the BWROG on Stability will occur. (LTCA #12)

27) Improve Effectiveness of Industry Involvement

Status: In-progress, an assessment by Quality Assurance/Engineering has been scoped and detailed planning is underway. (LTCA #14)