CATEGORY 1

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ACCESSION NBR:9610230085 DOC.DATE: 96/10/15 NOTARIZED: NO DOCKET #
FACIL:50-397 WPPSS Nuclear Project, Unit 2, Washington Public Powe 05000397
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Document Control Branch (Document Control Desk)

SUBJECT: Responds to NRC 960912 ltr re violations noted in insp rept 50-397/96-11.Corrective actions:Design Engineering Manager approval of all TER changes required.On 960724 scaffolding stockpiles in RCIC & RHR "A" & "B" rooms removed during insp

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October 15, 1996
GO2-96-201

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21**
NRC INSPECTION REPORT 96-11, RESPONSE
TO NOTICE OF VIOLATIONS

Reference: Letter dated September 12, 1996, KE Brockman (NRC) to JV Parrish (SS), "NRC Inspection Report 50-397/96-11 and Notice of Violation"

The Supply System's response to the referenced Notice of Violations, pursuant to the provisions of Section 2.201, Title 10, Code of Federal Regulations, is enclosed as Attachment A. Enclosed as Attachment B are additional comments regarding staff findings noted in Inspection Report 96-11.

Should you have any questions or desire additional information regarding this matter, please call me or Ms. L. C. Fernandez at (509) 377-4147.

Respectfully,

P. R. Bemis
Vice President, Nuclear Operations
Mail Drop PE23

Attachments

IE01 1/1

cc: LJ Callan - NRC RIV
KE Perkins, Jr. - NRC RIV, Walnut Creek Field Office
NS Reynolds - Winston & Strawn
TG Colburn - NRR
DL Williams - BPA/399
NRC Sr. Resident Inspector - 927N

The executive summary for the inspection report stated that, "The permanent plant modification program was not effectively implemented due to...." In subsequent discussion with Mr. Tom Stetka and Mr. Ken Brockman the Supply System requested, and your staff members concurred, that this statement does not correctly characterize the Supply System's implementation of the modification program. We request that the executive summary be changed to convey a clear message that the Supply System's permanent modification program is effectively implemented. However, some problems existed with the equivalent change process which must be addressed.

VIOLATION A

RESTATEMENT OF VIOLATION

10 CFR Part 50, Appendix B, Criterion V, states, in part, that activities affecting quality shall be accomplished in accordance with instructions, procedures, or drawings.

Plant Procedures Manual 1.4.1, "Plant Modifications," Revision 22, Section 5.1, Action 10, required, in part, that if the recommended action of a technical evaluation request was an equivalent change then changes could be made in accordance with Technical Services Instruction TI 1.2, "Equivalent Change Evaluations."

Technical Services Instruction TI 1.2 "Equivalent Change Evaluations," Revision 1, Section 4.5, required, in part, that the equivalent change process not be used for complex plant modifications or where there were significant impacts to existing calculations.

Plant Procedures Manual 10.2.53, "Seismic Requirements for Scaffolding, Ladders, Man-Lifts, Tool Gang Boxes, Hoists, and Metal Storage Cabinets," Revision 14, required, in part, that all scaffolding be left in an acceptable seismic configuration and, if it did not meet procedural requirements, that an engineering evaluation be performed. In addition, Section 7.1.6, required, in part, that for partially erected or removed scaffolding, an engineering evaluation, which included a 10 CFR 50.59 safety evaluation, be performed.

Contrary to the above:

1. As of July 22, 1996, Technical Evaluation Requests 94-0264, 94-0306, and 96-0004 performed plant modifications using the equivalent change process even though the modifications were complex or involved significant impacts to existing calculations.

2. On July 10, 1996, an engineering evaluation had not been performed for removed scaffolding, which was stored in an unacceptable seismic configuration near reactor core isolation cooling Pump P-1 and residual heat removal Pump 2C.

These examples, in aggregate, constitute a Severity Level IV violation (Supplement I) (50-397/9611-01).

RESPONSE TO VIOLATION A-1

The Supply System accepts this violation.

REASON FOR VIOLATION

The Supply System acknowledges that weaknesses do exist in the procedures and instructions that govern the Technical Evaluation Request (TER) equivalent change process. The Supply System does not feel that the implementation of plant modifications using the equivalent change process for TERs 94-0264, 94-0306, and 96-0004 was "a failure to adhere to plant engineering and maintenance procedures," as was noted in Inspection Report 50-397/96-11. The Supply System adhered to the appropriate plant procedure and concluded that implementing the TERs as equivalent changes was appropriate.

As noted by the staff, Plant Procedures Manual (PPM) 1.4.1, "Plant Modifications," Revision 22 is the governing procedure for the implementation of permanent plant modifications. As also noted, PPM 1.4.1 allows TERs to implement permanent plant modifications which are considered to be equivalent changes. PPM 1.4.1, and Technical Instruction (TI) 1.2, "Equivalent Change Evaluations," define an equivalent change as "A hardware change that results in installation or modification of an item or items, not identical to the original item or items, which meets the design bases of both (i) the item(s), and (ii) applicable interfaces." This definition was taken from the Electric Power Research Institute (EPRI) "Guidelines for Optimizing the Engineering Change Process for Nuclear Power Plants," March, 1994. Section 5.1 of PPM 1.4.1, item 10 notes that if the recommended action to resolve a TER is an equivalent change, TI 1.2 should be used to provide an equivalent change evaluation. TI 1.2 does provide guidance and direction for performing the equivalent change evaluation, but as noted in Section 1.0 of TI 1.2, the TI 1.2 guidance is supplemental to PPM 1.4.1. PPM 1.4.1, Section 5.1, item 10 allows the design discipline engineering manager to approve the equivalent change evaluation and the equivalent change, provided the change is:

- Beneath the level of detail in the design documents; or
- Within the level of detail contained in design documents but of configuration nature only; and
- Within the existing design basis and design requirements.

TI 1.2 also provides additional guidelines to aid in the performance of an equivalent change evaluation. As noted in Section 2.0 of TI 1.2, an equivalent change evaluation is performed to assure that:

- Technical requirements of the design basis function(s) of the original item(s) and applicable interfaces with other systems, structures, and components (SSCs) are met.

- The replacement item(s) will not affect the design bases function of the replaced item(s), or affect the design function of the interfacing SSCs.

- The replacement item(s) will not introduce any failure modes not considered in the current design bases.

TI 1.2 notes that an equivalency evaluation should describe and consider the functional description of the equipment, as well as all design basis requirements, such as seismic and environmental qualification, codes, standards, and regulatory requirements. An equivalency evaluation should also consider the interfaces with other SSCs, and provide a justification for the equivalent change.

The Supply System maintains that the three TER modifications, as noted by the staff in this violation to be "not equivalent," were implemented in accordance with approved plant procedures. The Supply System does acknowledge, however, that there are some ambiguities within the procedures that govern the equivalent change process that could add confusion as to when an equivalent change should or should not be done. Primarily, TI 1.2 lists in the "Guidelines" section of the instruction that "as a general rule, the equivalent change process should not be used for complex plant modifications or where there are significant impacts to calculations...." The Notice of Violation incorrectly stated that TI 1.2 listed this statement as a requirement. This statement does, however, allow for a subjective interpretation of the phrases "complex plant modifications" or "significant impacts to existing formal calculations." PPM 1.4.1, the governing procedure for the equivalent change process, provides more precise criteria for approving an equivalent change for implementation, as noted above.

The Supply System believes that the procedures that provide guidance and direction for the equivalent change process should be improved. These improvements can provide clear guidelines as to when the equivalent change process would be appropriate for implementing plant modifications.

CORRECTIVE STEPS TAKEN AND RESULTS ACHIEVED

As an interim action until procedures are revised, the Design Engineering Manager is now required to approve all TER equivalent changes to ensure they adhere to equivalent change guidelines and criteria. TERs 94-0264, 94-0306, and 96-0004 were reviewed on October 14, 1996, and it was determined that they were implemented in accordance with approved plant procedures.

CORRECTIVE STEPS TO BE TAKEN

The Supply System will revise PPM 1.4.1 and/or TI 1.2 to ensure consistent, clear, and conservative criteria are provided to determine when an equivalent change can be performed.

DATE WHEN FULL COMPLIANCE WAS ACHIEVED

Full compliance was achieved on October 14, 1996, when it was determined that TERs 94-0264, 94-0306, and 96-0004 were implemented in accordance with approved plant procedures.

RESPONSE TO VIOLATION A-2

The Supply System accepts this violation.

REASON FOR VIOLATION

The inspectors noted that scaffolding stockpiles or storage racks existed in the upper portion of the Reactor Core Isolation Cooling (RCIC) pump room and at the ground floor of the Residual Heat Removal (RHR) "C" pump room. PPM 10.2.53, "Seismic Requirements for Scaffolding, Ladders, Man-Lifts, Tool Gang Boxes, Hoists and Metal Storage Cabinets" requires, in part, that scaffolding be left in an acceptable seismic configuration as defined by the procedure. PPM 10.2.53 also requires that if scaffolding is left in a partially completed condition that does not meet the requirements of the PPM that an engineering evaluation be performed per Section 7.1.6 of this PPM. Section 7.1.6, in part, deals with partially erected/removed scaffolds which should be left in place and states that "Engineering should evaluate these specific scaffolds and conditions then provide acceptance/rejection. Engineering should complete an evaluation including a 50.59 review and construction criteria for each request as required." These stockpiles as well as one each in the RHR "A" and RHR "B" pump rooms were considered by NRC inspectors to be in an unacceptable seismic configuration, and per the NRC's interpretation of Section 7.1.6 of PPM 10.2.53, Revision 14, a 50.59 review was required for each occurrence.

The scaffolding stockpiles were requested by carpenters prior to installation, as required by PPM 10.2.53, in order to save exposure. Future work activities in these rooms would require the use of scaffolds and removing the scaffolding material from these rooms had the potential for accumulating unnecessary dose. Engineering was contacted and walkdowns were performed by a qualified structural engineer responsible for seismic evaluations. In addition, construction and stockpile restraint criteria was provided to preclude adverse impacts with safety-related systems or components during a seismic event. The stockpiles were then built per this criteria and inspected upon completion. This completed the engineering evaluation of these stockpiles per the responsible engineer's interpretation of the procedural requirements. Though no formal calculation was performed for these proposed stockpile structures the responsible engineer had performed the original analysis of scaffolds that formed the bases for PPM 10.2.53 with regard to erecting and restraining scaffolds for seismic concerns. The responsible engineer applied this

criteria and background knowledge to the proposed installations and determined that the proposed stockpiles would be seismically acceptable if they were installed as he had directed. A final walkdown was performed after the stockpiles were built and the final configuration was per the responsible engineer's direction. As noted above, PPM Section 7.1.6 contains both "should" and "as required" which was interpreted by the responsible engineer to provide him with the flexibility to use judgement as to when the evaluation required a written evaluation and a 50.59 review. The engineer compared the configurations of these stockpiles with acceptable erection criteria for scaffold configurations and determined that the stockpiles in the pump rooms were bounded by existing analyses and procedure requirements. Thus, the existing analyses and previously performed 10 CFR 50.59 safety evaluations for PPM 10.2.53 scaffold configurations was determined by the engineer to be bounding for these installations and no further documentation was required.

The staff has issued NOV 50-397/9611-01 to document a failure to comply with the procedural requirement of PPM 10.2.53 to perform an engineering evaluation for scaffold material stored in an unacceptable seismic configuration near safety-related equipment. This is based on the NRC interpretation of Section 7.1.6 of PPM 10.2.53 which in part, deals with partially erected/removed scaffolds which should be left in place and states that, "Engineering should evaluate these specific scaffolds and conditions then provide acceptance/rejection. Engineering should complete an evaluation including a 50.59 review and construction criteria for each request as required." The staff's interpretation in this case appears to be that "should" is a "shall' and that a "written" evaluation is required for each instance, and that this written evaluation "shall" include a 50.59 review. The responsible engineer interpreted the "should" in accordance with the definition in WNP-2 site wide procedure glossary and evaluated the scaffolds accordingly. This evaluation included a walkdown prior to erection of the subject stockpiles, verbal provision of construction criteria to the craft and walkdown of the final configuration to ensure craft compliance to verbal direction.

The primary reason for storage of the scaffolding material without a written evaluation and 10 CFR 50.59 safety evaluation is weakness in the wording of Section 7.1.6 of PPM 10.2.53. The wording of this section was ambiguous in that the one sentence contained both "should" and "as required." This wording left the intent of the section open to different interpretations including whether the requirement is really a "shall," and whether the engineering evaluation needs to be a written document.

CORRECTIVE STEPS TAKEN AND RESULTS ACHIEVED

The scaffolding stockpiles in the RCIC and RHR "A" and "B" rooms were removed during the NRC engineering inspection by July 24, 1996. The scaffolding stockpile in RHR "C" room was used to build a scaffold in the Low Pressure Core Spray (LPCS) pump room on July 25th, with all left over material and the stockpile structure removed on July 25, 1996.

CORRECTIVE STEPS TO BE TAKEN

The Supply System will revise PPM 10.2.53 to clarify that scaffolding stockpiles in safety-related areas will only be permitted when evaluated per the guidelines of PPM 10.2.53 and by a written 10 CFR 50.59 safety evaluation. Bounding 10 CFR 50.59 safety evaluations are considered an acceptable method of implementing this commitment. This procedure revision will be completed by November 29, 1996.

DATE WHEN FULL COMPLIANCE WAS ACHIEVED

Full compliance was achieved by July 25, 1996, when scaffolding material was removed from the RCIC and RHR pump rooms.

VIOLATION B

RESTATEMENT OF VIOLATION

10 CFR Part 50, Appendix B, Criterion III, requires, in part, that measures shall be established to assure that regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, as of July 24, 1996:

1. Technical Evaluation Request 94-0264 modified a standby liquid control system piping hanger from a welded to a bolted configuration without performing a design stress analysis for the currently installed bolted configuration.

2. Technical Evaluation Request 94-0306, which modified reactor core isolation cooling piping, had stress analyses performed for the use of all stainless steel piping and for all carbon steel piping, but not for the carbon steel/stainless steel combination that was installed. In addition, this request modified the pipe hanger configuration, but system drawings were not revised to reflect the actual installed configuration.

3. Technical Evaluation Request 96-0125 increased the clearances at the lever arm pivot of valve RCIC-V-2 without determining whether the clearances were consistent with vendor requirements.

These examples, in aggregate, constitute a Severity Level IV violation (Supplement I) (50-397/9611-03).

RESPONSE TO VIOLATION B

The Supply System accepts this violation.

REASON FOR VIOLATION

In regards to Violation B-1, a welded steel plate 1.0 x 5.0 x 0.5 inches was replaced with a bolted steel plate 6.0 x 5.0 x 0.5 inches, in order to simplify the replacement of Standby Liquid Control (SLC) squib valves. A stress analysis calculation had previously been performed for a proposed bolted configuration. However, the proposed bolted design was simplified at the time of actual field installation. A revision to the calculation was also prepared for the actual field installation. This revision, however, simply documented a review of the previous calculation and concluded that the modification was acceptable by extrapolation from the previous calculation. The Supply System acknowledges that the differences between the actual field installed configuration and the previous calculation were not clearly described in the revised calculation. Also, the revised calculation did not discuss why the previous calculation was conservative with respect to the field installation. During the NRC inspection, the revised calculation was again revised to reflect the actual field installation. This final effort reconfirmed that extrapolation from the previous calculation was in fact conservative and that the modification was acceptable.

In regards to Violation B-2, the staff reviewed a calculation that initially analyzed small bore, carbon steel, drain piping from the RCIC steam leg. This calculation had been revised earlier to document that a portion of the piping had been changed to stainless steel. This revision referred to a formal calculation that analyzed the entire length of piping as stainless steel. Again, the Supply System acknowledges that the revised calculation did not thoroughly document and describe how the revised calculation was bounded by the formal calculation. The Supply System maintains that the revised calculation was conservative because the routing, weight, and other pertinent parameters remained identical, and only the coefficient of thermal expansion was changed. Since stainless has a higher coefficient of thermal expansion, the bounding analysis for a combination of carbon/stainless was conservative. The intent was to replace the entire run of pipe with stainless, so the revised calculation was prepared to reflect this eventuality as well as bound the interim configuration. During the NRC inspection, the calculation was revised to more specifically reflect the as-built configuration of both carbon and stainless piping materials. The result confirmed that, when using carbon steel parameters, the stainless steel results were conservative as expected, and that the installation was acceptable as installed.

In a second issue as addressed in Violation B-2, the staff identified that drawings had not yet been revised to reflect modified pipe hanger configurations that resulted from TER 94-0306. The Supply System acknowledges this as a weakness in our design control process. The drawing in question was part of backlogged engineering work, and although marked up at the time of field work, the revisions had not yet been approved for incorporation into the station data base. The

significance of this finding was that the equivalent change procedure did not require tracking document changes early in the process, such that it could be determined when an outstanding change existed against a drawing.

Violation B-3 addressed the removal of a spacer washer that, when in place, prevented the crafts from restoring the RCIC-V-2 linkage clearances to that identified in the vendor's manual. These two small spacer washers were placed on the hinge pin between the clevis ears and the servo-actuator arm. In order to maintain the specified clearance, one spacer washer was removed. The clearance was then measured with feeler gauges and the actuator was physically stroked to assure that proper operation had been obtained. The actual work activity was monitored by Maintenance management, Design Engineering, System Engineering, turbine governor control experts, resident NRC inspectors, and experienced craft personnel. The Supply System acknowledges that documentation of vendor concurrence with removal of the washer was not included in the TER package. The manufacturer has since concurred with the removal of the spacer washer. The removal of the spacer washer was necessary to obtain vendor required clearances.

CORRECTIVE STEPS TO BE TAKEN

Revise design control procedures as necessary to ensure that:

1. The equivalent change process incorporates tracking of document changes early in the process, as does the normal design process, to ensure drawings are revised promptly to reflect plant modifications.

2. Clear guidance is provided for describing and justifying field configurations when bounding or extrapolated calculations are used.

3. The basis for design changes (including vendor concurrence when applicable) are adequately documented.

Training will also be provided to design and system engineering personnel to emphasize the importance of documenting the basis for design changes, including vendor concurrence when applicable.

DATE WHEN FULL COMPLIANCE WAS ACHIEVED

For violations B-1 and B-2, full compliance was achieved by July 26, 1996, when configuration calculations were revised and determined to be correct. For Violation B-3, full compliance was achieved when the vendor concurred with removal of the spacer washer on September 5, 1996.

VIOLATION C

RESTATEMENT OF VIOLATION

10 CFR Part 50, Appendix B, Criterion XVI, requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall ensure that the cause of the condition is determined and corrective action taken to preclude repetition.

Contrary to the above:

1. From January 19 to July 6, 1996, prompt and adequate corrective action for a condition adverse to quality was not taken. Specifically, on January 19, 1996, a failure of primary containment isolation valve FDR V-4 to close was identified to be caused by foreign material on the valve seating surfaces. However, this foreign material problem was not promptly corrected, and, as a result, additional failures of valve FDR V-4 and the redundant isolation valve, FDR V-3, were caused by this foreign material.

2. As of July 24, 1996, the licensee failed to take adequate corrective action to assure that the Corporate Nuclear Safety Review Board was receiving all of the 10 CFR 50.59 safety evaluations for review. As a result, Safety Evaluation 95-095 was not reviewed by the Corporate Nuclear Safety Review Board.

3. As of July 24, 1996, the licensee did not promptly correct a condition adverse to quality in that the required reading of operations personnel as a part of the corrective action for NRC Violation 50-397/9518-01, was not completed even though the violation had occurred over a year ago.

These examples, in aggregate, constitute a Severity Level IV violation (Supplement I) (50-397/9611-04).

RESPONSE TO VIOLATION C-1

The Supply System accepts this violation.

REASON FOR VIOLATION

The Supply System believes, based on the valve operating data available at the time FDR-V-4 failed to close on January 24, 1996, and based on data available at the time FDR-V-3 exceeded it's ASME leakage limit on April 26, 1996, that the corrective action which could have prevented the reoccurrence of failures of either FDR-V-3 or FDR-V-4 (i.e. flushing the FDR wetwell loop-seal piping) did not seem necessary given the information available at the time.

It was only after FDR-V-3 and FDR-V-4 demonstrated additional stroke time failures in July, 1996 that a high flow flush of the loop seal piping was determined to be the most appropriate corrective action.

On July 6, 1996, primary containment isolation valves FDR-V-3 and FDR-V-4 exceeded stroke time limitations and were declared inoperable. Subsequent flushing of the valve bodies of FDR-V-3 and FDR-V-4 using a demineralized water source resulted in stroke times within the acceptable range. As an additional means of verifying valve operability, local leak rate testing on FDR-V-3 and FDR-V-4 was performed on July 12, 1996. FDR-V-3 exceeded leakage limits while FDR-V-4 was well within the leakage limitations. The FDR-V-3 ball and seat assemblies were dissembled, and it was discovered that the seat assemblies were covered in some places with a thin layer of gritty debris. The ball and seat assemblies were cleaned and inspected, and the valve reassembled. Both FDR-V-3 and V-4 were declared operable on July 18, 1996 based on successful completion of leakage and stroke time testing.

Two events associated with FDR-V-3 and FDR-V-4 occurred prior to the FDR-V-3 and FDR-V-4 slow stroke time events of July, 1996. On January 24, 1996, FDR-V-4 failed to indicate closed while performing surveillance testing. Investigations showed that the problem was not with position indication and that the valve was only moving to the intermediate position. The valve was cycled several times and finally moved full stroke, giving the proper indication. The valve was declared inoperable. Subsequent flushing of the valve using a demineralized water source resulted in stroke times within the acceptable range, and the valve being declared operable. On April 26, 1996, FDR-V-3 failed it's ASME leakage rate limit. The ASME limits for FDR-V-3 are 442 sccm and the as-found leakage was 742.5 sccm. Operability of FDR-V-3 was demonstrated following replacement of the valve seat assemblies and a rebuild of the air actuator. The failure of FDR-V-4 to close on January 24, 1996 was the first operational failure with this valve since its installation in 1991. The initial troubleshooting effort focused on two potential problem areas: 1) problems internal to the valve, and 2) problems external to the valve (i.e. the air actuator). In an effort to identify the problem without performing intrusive examinations of the internals of the valve or its actuator, a flushing activity was performed in order to determine if debris on the valve seat assemblies was the cause of the failure. Based on the improved valve performance and observation of debris exiting the downstream piping following the flush, it was determined that debris was the most likely cause of the failure. As a result, no intrusive inspections were deemed necessary at that time, and the debris was attributed to a gradual accumulation in the valve since its installation nearly five years earlier. Had an intrusive inspection of FDR-V-4 (and possibly FDR-V-3 as a result of the generic implication) been performed at that time, it may have provided solid physical evidence to support the debris theory, but it is believed it would have provided <u>no evidence</u> to suggest the debris accumulation was due to anything but five years worth of normal operating conditions.

When FDR-V-3 failed its leak rate testing on April 26, 1996, the discovery of debris on the valve seat assemblies supported the earlier determination that debris had the potential for affecting the operability of FDR-V-3 and FDR-V-4. Because this was the first operational failure of FDR-V-3 since its installation, this failure was also attributed to a gradual

accumulation of debris since installation of the valve in 1991. The fact that no significant amounts of debris were found in the upstream FDR drywell sump during a 1996 refueling outage inspection reinforced the assumption of a gradual deposition mechanism since there was no evidence of an increase (i.e. step change) in the amount of debris in the FDR piping over levels previously observed.

The reoccurrence of failures in FDR-V-3 and FDR-V-4 in July, 1996, provided operational history to establish debris deposition as something other than gradual in nature. After the discovery of significant debris buildup on the seat assemblies of FDR-V-3 in July, just months after their replacement in April, 1996, evidence became available to support a short-term debris accumulation mechanism. After identifying that such a short-term mechanism was at work, it became apparent that the probable source of the debris was accumulation in the FDR wetwell loop-seal piping. This debris accumulation apparently has reached a point where it now has the potential to affect the operability of FDR-V-3 and FDR-V-4.

The Supply System maintains that given the information available at the time, one would not necessarily conclude, as stated in the NOV, that because of a single failure of FDR-V-4 to stroke in January, 1996, that flushing was the most appropriate and necessary corrective action. The corrective actions initiated were considered prudent and appropriate based on the information available at the time. Until 1996, both FDR-V-3 and FDR-V-4 had operated without incident for 5 years. It was not until the FDR-V-3 and FDR-V-4 failures in July, 1996, that the probable source of debris that was affecting valve operability became evident.

CORRECTIVE STEPS TAKEN AND RESULTS ACHIEVED

The frequency of performing FDR valve operability stroke time testing has been increased from quarterly to weekly. This increased frequency will continue until future corrective actions associated with FDR-V-3 and FDR-V-4 are completed, or until an engineering evaluation of the stroke time data justifies decreasing the test frequency.

CORRECTIVE STEPS TO BE TAKEN

The Supply System will prepare and implement a plant modification during the 1997 refueling outage which will allow a high flow flush to be performed on the piping between the FDR drywell sump and valves FDR-V-3 and FDR-V-4. The Supply System will also perform a high flow flush during the 1997 refueling outage on the piping between the FDR drywell sump and valves FDR-V-3 and FDR-V-4.

DATE WHEN FULL COMPLIANCE WAS ACHIEVED

Full compliance was achieved by July 18, 1996, when FDR-V-3 and FDR-V-4 were returned to operable status.

RESPONSE TO VIOLATION C-2

The Supply System denies this violation.

REASON FOR VIOLATION DENIAL

During NRC Inspection 96-11, the NRC performed a review of 10 CFR 50.59 Safety Evaluations (SEs). One aspect of the review was to verify that the Corporate Nuclear Safety Review Board (CNSRB) was receiving and reviewing SEs, as required by Technical Specification 6.5.2.7.a. As a result of the NRC review, one SE was cited for not being sent to CNSRB for review (SE 95-095). The basis for this conclusion was that this SE was not listed on Attachment 5 of the CNSRB meeting minutes. Attachment 5 identified the Safety Evaluations reviewed for CNSRB meeting 96-05.

The NRC review was performed using information provided by the Supply System. Specifically, Plant Operations Committee (POC) meeting minutes, CNSRB meeting minutes and the SE number log were provided. Based on the information provided, the NRC concluded that SE 95-095 had not been sent to the CNSRB for review. This information was identified as an example of inadequate corrective action. The Supply System has since performed a review and determined that SE 95-095 had been reviewed by the CNSRB at the May 1-2, 1996 CNSRB meeting (Mtg#96-05). A typographical error in Attachment 5 to the meeting minutes incorrectly identified SE 95-062 instead of SE 95-095 as having been reviewed, thus leading the staff to conclude the SE 95-095 had not been reviewed. The Supply System has since reviewed the SEs sent to the CNSRB for meeting 96-05 and these were observed to contain SE 95-095, and not SE 95-062. In addition, based on the subject listed for what was identified in Attachment 5 of the meeting minutes as SE 95-062 (which should have been SE 95-095), this item was indeed SE 95-095. The subject of SE 95-095 was a change to the FSAR to reflect a criticality analysis performed for ABB fuel and to reflect changes in the fuel handling process.

CORRECTIVE STEPS TO BE TAKEN

The typographical error in CNSRB #96-05 meeting minutes will be corrected to indicate that SE 95-095 was reviewed by the CNSRB. This corrective action will be completed by November 29, 1996.

RESPONSE TO VIOLATION C-3

The Supply System accepts this violation.

REASON FOR VIOLATION

In response to Notice of Violation 50-397/9518-01, the Supply System had committed to the staff to include the violation response in required reading for licensed and non-licensed operators, and to have the event discussed with Operations crews. The violation involved the use of fire

protection water for non-fire protection activities. The event was discussed with Operations crews, and the response was included as part of Operations Department required reading. However, as noted in Inspection Report 50-397/96-11, the Supply System's corrective actions were not completed because the required reading was completed by only about 50% of Operations Department personnel.

The commitment to the staff to include the violation response in Operations Department required reading was being tracked by the Supply System's Plant Tracking Log (PTL). An Operations Department person had been assigned responsibility for follow-up and closure of the required reading action item in PTL. Operations management expectations were that the PTL item not be closed out until all appropriate Operations personnel had read the violation response. This person assumed that because the violation response was included in Operations required reading, as was committed to, the PTL item could be closed out as having been completed. Accordingly, the PTL item was closed out and no longer tracked by the Operations Department, even though all personnel had not performed the required reading.

CORRECTIVE STEPS TAKEN AND RESULTS ACHIEVED

Operations Department licensed and non-licensed personnel have completed the required reading regarding the misuse of fire protection water. Operations management has counseled the individual involved concerning expectations on PTL item closure when it is related to Operations Department required reading. Operations management has also addressed this incident with Operations personnel via a Night Order, to emphasize the importance of required reading and the expectation of completion in a reasonable time period.

DATE WHEN FULL COMPLIANCE WAS ACHIEVED

Full compliance was achieved when required reading was verified to be complete on August 21, 1996.

VIOLATION D

RESTATEMENT OF VIOLATION

Technical Specification 6.8.1 requires, in part, that written procedures be established to implement the requirements of NUREG-0737.

Contrary to the above, as of March 7, 1996, a requirement of NUREG 0737 to have an Independent Safety Engineering Group, did not have an implementing procedure describing the responsibilities and functions of the Nuclear Safety Assurance Division (the licensee's equivalent of the Independent Safety Engineering Group).

This is a Severity Level IV violation (Supplement IV) (50-397/9611-05).

RESPONSE TO VIOLATION D

The Supply System accepts this violation.

REASON FOR VIOLATION

The Supply System did have an implementing procedure (PPM 1.10.8, "Nuclear Safety Assurance Assessments") describing the responsibilities and functions of the Nuclear Safety Assurance Division (NSAD). PPM 1.10.8 satisfied Technical Specification (TS) 6.8.1 and 6.8.2 requirements by providing a formal procedure reviewed by the Plant Operations Committee (POC) and approved by the Plant General Manager describing the NSAD functions. PPM 1.10.8 was cancelled in 1993 due to the procedure being a restatement of the requirements located in Nuclear Operating Standards 20, "Quality Assurance Evaluations." The 10 CFR 50.59 evaluation contained in the procedure revision package for the deletion of PPM 1.10.8 failed to address the TS 6.8.1 requirement regarding procedures required to implement NUREG-0737 requirements.

CORRECTIVE STEPS TAKEN AND RESULTS ACHIEVED

PPM 1.10.8 has been reactivated and is now a POC reviewed and Plant General Manager approved procedure describing the NSAD functions.

Quality Services personnel have been trained regarding this event and its applicability to the processing of future procedure revisions.

CORRECTIVE STEPS TO BE TAKEN

The Quality Services Department will perform a surveillance to assess Plant compliance with TS Section 6 procedure requirements, to include procedural implementation requirements for NUREG-0737. This surveillance will be performed by February 16, 1997. In addition, a surveillance of past procedure cancellations will be performed by October 31, 1996 to ensure the requirements of TS Section 6 were considered.

DATE WHEN FULL COMPLIANCE WAS ACHIEVED

Full compliance was achieved on September 26, 1996, when the Plant General Manager approved PPM 1.10.8.

The following additional information is provided in response to comments made by the staff in Inspection Report 50-397/96-11:

STAFF COMMENT

The staff noted in Inspection Report Section E1.1.4.b that there was inconsistency in the FSAR as to whether the Reactor Core Isolation Cooling (RCIC) System was safety-related and seismic Category I.

RESPONSE TO COMMENT

The Supply System acknowledges that FSAR inconsistencies exist with respect to the safety classification of the RCIC System. The FSAR does note in Section 7.5.2.2.3 that the RCIC System does not perform a safety function. The FSAR also notes in NRC Question Section 013.136 that the RCIC System is not considered safety-related. However, FSAR Table 3.2-1, which summarized the safety classification of various RCIC System components, erroneously indicated that certain non-safety-related portions of the RCIC were Quality and Seismic Class 1, and therefore safety-related. An amendment to the FSAR is being processed to correct the RCIC informational discrepancies.

The RCIC System is not part of the Emergency Core Cooling Systems (ECCS), nor is it required to function following a loss of coolant accident (LOCA). The RCIC System can, however, be used to accomplish a function similar to that of an Engineered Safety Feature (ESF) System. The RCIC System is designed to operate either automatically or manually to provide core cooling and control of RPV water level. Under these conditions, the High Pressure Core Spray (HPCS) and RCIC systems perform similar functions.

The RCIC System consists of a steam-turbine driven pump and associated valves and piping capable of delivering water to the reactor vessel. The RCIC System initiates automatically on low reactor pressure vessel level, or can be initiated manually by a control room operator. The turbine is driven by reactor steam. Water is taken either from the condensate storage tank (CST) or the suppression pool and delivered to the reactor. Turbine exhaust is directed to the suppression pool where it is condensed. Alternate discharge flow paths are provided to allow recirculation to the CST for testing purposes.

Although the RCIC System was originally designed as a safety-related system at WNP-2, it was downgraded to non-safety related in 1985 when modifications were made to the Automatic Depressurization System (ADS) which enveloped the safety functions then performed by RCIC. The Supply System committed, in 1983, to perform the ADS modifications, and informed the staff at that time that the ADS modifications would result in the RCIC System being removed from the Equipment Qualification Program (see References 2 and 3 below). Staff approval of

the ADS modifications was provided by Amendment No. 11 to the Technical Specifications (see Reference 1 below). The ADS modifications consisted of removing ADS high drywell instrumentation and adding manual inhibit switches to the ADS logic. The WNP-2 FSAR was also revised in 1985 to reflect the RCIC classification downgrade.

Portions of the RCIC System remain classified as safety-related because of interfaces with other safety-related structures, systems, and components (SSCs). There are safety-related valves that form a boundary for the RCIC System, which is considered a closed system outside of containment. As a closed system outside of containment, RCIC is credited as the second barrier for containment isolation for the two RCIC lines that penetrate containment, each of which has only a single containment isolation valve. The portion of RCIC that forms a closed system outside of containment primarily extends from the single suppression pool suction line containment isolation valve to the injection line outboard containment/reactor coolant pressure boundary valve. Additionally, the minimum flow branch line from the main RCIC pump discharge line to the suppression pool has a single primary containment isolation valve. Piping between these valves form the main loop portion of the closed system outside of primary containment. There are also safety-related RCIC System boundary valves that interface with primary containment and serve as primary containment isolation valves, such as the steam supply valves to the RCIC turbine, the isolation valves on the RCIC turbine exhaust line to the suppression pool, and the isolation valves on the RCIC vacuum pump discharge line to the suppression pool.

There are also safety-related electrical components within the RCIC System. The RCIC motor-operated isolation and primary containment isolation valves discussed above receive safety-related power to ensure they will isolate on demand. The RCIC leak detection is safety-related and therefore uses a safety-related 24 volt power supply. In addition, RCIC primary containment isolation valve control room position indication is a Regulatory Guide 1.97, Category I requirement, and is therefore a safety-related circuit.

The RCIC System also supports certain augmented quality functions. An augmented quality function is a Supply System designation for SSCs that are <u>not</u> safety-related, but are subject to special design and/or quality assurance requirements. RCIC is credited for maintaining reactor vessel water level during an Anticipated Transient Without Scram (ATWS) event. Therefore, many of the RCIC components receive the ATWS augmented quality designation. Many of the non-safety-related RCIC components that would normally be designated as Seismic Category II, are located near Seismic Category I equipment in areas such as primary containment, and the reactor and radwaste buildings. These components have been evaluated and assigned a seismic augmented quality designator. During a seismic event these Category II/I augmented quality SSCs are designed to not fail in a manner to render safety-related seismic Category I components inoperable. Selected RCIC instrumentation and controls are duplicated on the remote shutdown panel to allow monitoring and control of RCIC if it is necessary to evacuate the main control room. Therefore, the RCIC System is credited with supporting remote shutdown and assigned

the remote shutdown augmented quality designator. Also, WNP-2 credits RCIC to be available during a Station Blackout event in order to provide defense in depth. (High Pressure Core Spray is the core make-up system for Station Blackout). Therefore, the Station Blackout augmented quality designator is assigned to the RCIC System.

In summary, except for those portions of the RCIC System that interface with other safety-related SSCs, RCIC serves a non-safety related function and is not credited to mitigate design basis accidents. The Supply System, however, does acknowledge that Probabilistic Safety Assessment has generally shown that the availability of the RCIC System contributes to an overall reduction in risk to public health and safety. RCIC is presently addressed in the WNP-2 Technical Specifications and, in accordance with Criterion 4 of 10 CFR 50.36, RCIC System operability will be maintained in the Improved Technical Specifications.

Reference:

1) Letter, GI2-85-98, dated July 1, 1985, WR Butler (NRC) to GC Sorensen (SS), "Issuance of Amendment No. 11 to Facility Operating License NPF-21, WPPSS Nuclear Project No. 2"

2) Letter, GO2-83-660, dated July 26, 1983, GC Sorensen (SS) to NRC, "Safety Evaluation Report, NUREG-0892, Outstanding Issue 1.7(9), Modifications to ADS System Logic"

3) Letter, GO2-83-408, dated May 4, 1983, GD Bouchey (SS) to A Schwencer (NRR), "Safety Evaluation Report, NUREG-0892, Outstanding Issue 1.7(9), Modifications to ADS System Logic"

STAFF COMMENT

The staff noted in Inspection Report Section E2.2 that Safety Evaluation 95-025 "was written for relocation of the postulated break for Standby Liquid Control System piping. While the safety evaluation acknowledged that the relocation of the break may cause some equipment located in the vicinity of the break to be damaged, the effect of this damage was not further evaluated other than concluding that the plant would still be able to safely shutdown."

RESPONSE TO COMMENT

The text contained in Safety Evaluation 95-025 did not specifically identify the equipment that may be damaged if the postulated break were to occur. However, in the reference section of the 10 CFR 50.59 screening review for Safety Evaluation 95-025 the following Calculation Modifications Records (CMRs) are listed: CMR-94-1192 (against calculation number 5.49.50), CMR-94-1193 (against 5.49.51), and CMR-94-1194 (against 5.49.52). CMR-94-1192

documents the behavior of the postulated pipe break should it occur. CMR-94-1193 identifies the specific targets that would be impacted if the postulated break were to occur. CMR-94-1194 evaluates the specific effect of the damage to the targets that were identified. The conclusion of CMR-94-1194 is that the equipment damaged, if this postulated break were to occur, would not affect the ability of the plant to achieve a safe shutdown condition. Safety Evaluation 95-025 (by way of reference to CMR-94-1194) does include an evaluation of the effect of the damage to the individual equipment located in the vicinity of the postulated break on the RPV side of SLC-V-8.

STAFF COMMENT

In Inspection Report Section E3, the staff noted "two examples during the review of Procedure 1.3.43 where the procedural requirements were inconsistent with the requirements of 10 CFR 50.59. In the first example, Procedure 1.3.43 allowed the Plant Operations Committee to recommend implementation of activities that involve a Technical Specification change or an unreviewed safety question prior to NRC approval so long as the equipment is not declared operable or relied upon for nuclear safety until NRC approval is obtained. The team noted that 10 CFR 50.59 makes no provision for this exception, however, the team did not identify any examples where application of this exception resulted in a safety concern."

In the second example, the guidance for conducting 10 CFR 50.59 safety evaluations allowed that an accident of a new or different type than any accident previously evaluated need not be considered if the likelihood of that accident is less than that of accidents previously evaluated. Again, the team noted that 10 CFR 50.59 did not provide for this exception and stated that an unreviewed safety question exists if the possibility of a new or different type of accident is created, however, the team did not identify any examples where application of this exception resulted in a safety concern.

RESPONSE TO COMMENTS

With respect to the first example, the Supply System does not believe that this policy is contrary to 10 CFR 50.59. A decision to begin implementation of an activity that involves a Technical Specification change or an unreviewed safety question is a business risk decision and outside the scope of 10 CFR 50.59 criteria. Clearly, the equipment could and would not be declared operable or relied upon for nuclear safety without prior staff approval. In addition, the installation work is not allowed to impact the operability of other equipment. Implementation of activities that involve Technical Specification changes or an unreviewed safety question prior to NRC approval is the exception rather than the rule. For these cases, every effort is made to communicate the specifics of the implementation plan to the staff to ensure a mutually acceptable review and approval schedule.

With respect to the second example, the guidance provided in PPM 1.3.43 for determining whether an accident is a new or different type than any previously evaluated in the safety analysis report is consistent with the guidance provided in NSAC-125. We realize that the NRC has not endorsed NSAC-125. But in recently issued NRC inspection guidance and other industry correspondence, the two outstanding issues identified relative to NSAC-125 criteria are related to "slight increases in probability" and "margin of safety." It has been and continues to be our understanding that the NRC has not previously had a concern with the NSAC-125 guidance regarding new and different types of accidents. We will evaluate changes to our process when the outstanding issues are resolved between the Nuclear Energy Institute and the NRC and additional guidance is available.

For these reasons, we do not believe our procedural guidance is inconsistent with 10 CFR 50.59 requirements.

STAFF COMMENT

In Inspection Report Section E2.2, the staff noted during the review of safety evaluations that many of the forms were outdated.

RESPONSE TO COMMENT

It is a Supply System policy that if a safety evaluation had been initiated prior to a new form being issued, the outdated form is allowed to continue through the process. Previous forms were in full compliance with the regulation at the time of use. There is no requirement that the Safety Evaluation be redone on the new form. To impose such a requirement would impose an unnecessary administrative burden with no quality improvement. Accordingly the practice of accepting outdated forms will be maintained.