UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report Nos.: 50-390/86-20 and 50-391/86-20

Licensee: Tennessee Valley Authority
6N11 B Missionary Place
1101 Market Street
Chattanooga, TN 37402-2801

Docket Nos.: 50-390 and 50-391 License Nos.: CPPR-91 and CPPR-92

Facility Name: Watts Bar 1 and 2

Inspection Conducted: August 21 - September 20, 1986

Inspectors: GK Afuneys FOR December 30, 1986
M. W. Branch, Senior Resident Inspector Date Signed
Operations

GK Afuneys FOR December 30, 1986
G. A. Walton, Senior Resident Inspector Date Signed
Construction

GK Afuneys FOR December 30, 1986
P. G. Humphrey, Resident Inspector Date Signed

GK Afuneys FOR December 30, 1986
W. Bearden, Resident Inspector Date Signed
Bellefonte (September 8-12, 1986)

Approved by: GK Afuneys FOR December 30, 1986
M. B. Shymlock, Section Chief Date Signed
Division of TVA Projects

SUMMARY

Scope: This routine inspection was conducted in the areas of licensee action on previous enforcement items, licensee action on inspector identified items, followup of licensee identified items, fire prevention and fire protection, preoperational test (PREOPS) program implementation verification, testing of pipe support and restraint systems, other safety related piping, reactor vessel and internals work observation, structural welding, in-depth Quality Assurance (QA) inspection of performance, and verification of as-builts.

Results: One violation, two unresolved items and one inspector followup item were identified in this report.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*G. Toto, Site Director
D. M. Lake, Construction Project Manager
*R. A. Pedde, Unit 2 Nuclear Project Manager
*E. R. Ennis, Plant Manager
R. C. Parker, Site Quality Assurance Manager
*J. A. McDonald, Licensing Manager
*R. C. Miles, Modifications Manager
*H. B. Bounds, Maintenance Superintendent
B. S. Willis, Operations and Engineering Superintendent
B. F. Painter, WBN Construction
*J. P. Mulkey, Quality Assurance Supervisor
*H. C. Johnson, WBN Quality Assurance
R. Norman Jr., Acting Operations and Engineering Superintendent
R. D. Tolley, Design Services Manager
J. S. Woods, Instrument Maintenance Supervisor
J. L. Collins, Mechanical Maintenance Supervisor
*M. K. Jones, Engineering Group Supervisor
H. M. De Souza, Electrical Maintenance Supervisor
*R. R. Garu, Preoperational Test Section Supervisor
*L. E. Ottinger, Plant Compliance Staff, Nuclear Engineer
C. A. Borelli, Plant Compliance Staff, Nuclear Engineer
R. L. McKnight, Projects Engineer, Design Services
M. E. Reeves, WBN Project Engineering

Other licensee employees contacted included engineers, technicians, nuclear power supervisors, and construction supervisors.

*Attended exit interview

2. Exit Interview

The inspection scope and findings were summarized on September 22, 1986, with those persons indicated by an asterisk in paragraph one above. The following new items were discussed:

- Violation 390/86-20-01; Failure to properly change Plant Operation Review Committee (PORC) approved procedures (paragraph 3.a).

- Unresolved Item 390/86-20-02; Review of Post Modification Testing (paragraph 5.c).

- Unresolved Item 390/86-20-03 and 391/86-20-01; Review of Seismic Qualification Reports for Class IE equipment (paragraph 14).

- Inspector Followup Item 390/86-20-04 and 391/86-20-02; Quality Control (QC) inspector's access and use of QC inspection procedures (paragraph 13).

The licensee acknowledged the inspection findings with no dissenting comments. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection period. At no time during the inspection period did the inspectors provide written material to the licensee.

3. Licensee Action on Previous Enforcement Items (92702)

   a. (Closed) URI 390/86-16-01; Method Used to Change Plant Operations Review Committee (PORC) approved procedures. In inspection report 390/86-16, the inspector identified a condition where PORC approved procedures were modified without a formal change request being processed. Specifically, test and retest of plant safety related equipment per test addenda, were handled informally without proper review or approval. Additionally, PORC approved procedures were being partially completed and some steps, determined by the user to be unnecessary, were marked as not applicable (N/A). Both of the above described practices appeared to circumvent the formal change process.

      Inspection report 390, 391/86-16 described the procedure change process which is repeated below for clarity:

      - Quality Assurance (QA) Topical Report section 17.2.2.1, defines the scope of the Operational Quality Assurance Program (OQAP) to include activities being conducted after system turnover from the construction organization. This section also specifically defined preoperational (PREOP) testing as an activity covered by the OQAP program.

      - Section 17.2.5, "Instruction, Procedures and Drawing", of the QA Topical Report requires that procedures affecting the safety related functions of Critical Systems, Structures and Components (CSSC), be PORC reviewed.

      - Section 17.2.11, "Test Control", of the QA Topical Report requires that PREOP testing be accomplished in accordance with written and approved procedures.

      - The QA Topical Report also states that retests are to be conducted in the same manner as the original tests.

      - ANSI 18.7-1976, Section 5.2.2, which is committed to in QA Topical Report Table 17.D-3, requires that approved procedures be followed. However, changes to these procedures may be made without involving the original review/approval process provided the change clearly does not change the intent of the original procedure.

- Watts Bar Administrative Instruction (AI) 3.1, revision 15, "Plant Instructions - Control and Use", requires intent changes be handled through the same formal review process as the original issued instruction. The procedure defines intent change as a change which goes beyond the intent of the original instruction. However, the definition is restricted to the following:

  - Changes to plant instructions acceptance criteria.
  - Deletion or alteration of Quality Control (QC) hold points. Instruction changes affecting QC hold points may be handled as nonintent if documented Plant Quality Assurance (PQA) concurrence is obtained prior to implementing the change. Addition of new hold points does not constitute an intent change.
  - Changes to instructions which would violate the Technical Specifications or other licensing requirements.
  - Changes which increase the probability or consequences of equipment malfunctions or accidents.
  - Changes in scope, technique, or sequential order of instruction steps that would affect the result or nuclear safety.
  - Changes which implement a temporary alteration (TA) to operable CSSC without a TA Control Form. For example, if a change to a workplan called for temporary removal of an interfering hanger on an operable CSSC system, the change must be handled as intent to assure a before-the-fact PORC review.
  - Changes to either the authority or responsibility for review and/or approval of the document or the results obtained by its implementation.
  - In addition to the above, Final Safety Analysis Report (FSAR) requirements contained in chapter 14 section 14.2.4.1., titled, "Conduct of the Preoperational Test Program", specifies that "During performance of the test, minor changes to the test instruction not involving safety-related aspects and not interfering with test objectives or invalidating test results may be documented and reported to the test program coordinator. Major or significant changes to a test instruction are subject to the same review and approval as the original instruction".

Because of the complexity of many logic circuits and their significance to safety, it is essential that the system be tested to ensure that operating parameters meet design criteria and that on the spot test changes are not made to accommodate as-constructed conditions.

For this reason, a proper and formal review of changes to safety-related test procedures is required.

The inspectors reviewed several test addendum packages as well as several completed Unit 1 preoperational tests to determine if changes to PORC approved procedures were properly handled.

Completed preoperational test procedures reviewed include: W-2.1 (Chemical and Volume Control System); W-3.1 (Safety Injection System); and W-4.1 (Residual Heat Removal System).

Several test changes associated with these system test packages were found to conflict with the FSAR requirements. Specific examples where significant test changes were treated as non-significant are listed below:

Test #	Change #		Description of Changes
W-2.1	3, 4, 5, 6, 7, 8, 15, 16, 18, 22	-	Changing controller position to achieve the desired indication
		-	Changing required test result from 'continuity' to indicate 'discontinuity'
		-	Adding jumpers to achieve different signals
		-	Revising test method
		-	Changing volts to milli-amps
		-	Changing valve position from close to open, and changing the testing logic
W-3.1	2	-	Eliminates step
	4 and 7	-	Adds steps to test
	6	-	Changes calibration curves
	10	-	Allows test to be performed out of sequence at the direction of the test director
W-4.1	1,2,3,4,& 5	-	Changes in logic

The above deficiencies occurred during the 1983 time frame and are additional examples of improper test changes which were the subject of Violation 390/83-32. TVA feels that corrective action specified for that violation should correct these types of deficiencies.

In addition to the above completed tests, the inspector reviewed several test addendum packages. This review was performed to determine if retesting of equipment, which either failed the initial test or was bypassed as a result of equipment availability was performed in accordance with PORC approved procedures. The test addenda reviewed, along with the inspector's findings, are listed below:

- Retest of Deficiency Notice (DN) -1 to preoperational test TVA-25B, "High Pressure Fire Pumps (HPFP) Water Supply System." This retest was not PORC approved and consisted of performing only selected sections of the original test. Additionally, the specified retest, approved on February 5, 1985, referenced a special test equipment section from the original test which did not reflect change 2, dated February 14, 1984, to the original test. This change modified the special test equipment list.

- Retest of Exception Notice -11 for preoperational test TVA-9A, "Auxiliary Building Gas Treatment System (ABGTS)." This addendum was not PORC approved and only repeated selected sections of the original test without repeating all applicable prerequisites and precautions. This retest was completed on April 4, 1984.

- Retest of DN-22 to preoperational test W-3.1F, "Integrated Engineered Safety Features Actuation." This retest was completed and accepted without a PORC approved test procedure. Written procedure steps were not utilized nor were selected steps from the original procedure repeated. This retest was performed on May 1, 1985, by performing activities not referenced within the scope of the W-3.1F test procedure.

The above test addenda modify the scope, technique or sequential order of the original test procedure and therefore constitute an intent change to the original test. The use of test addenda as described above circumvent the required review/approval process and violate QA Topical Commitments necessary to ensure FSAR requirements are satisfied. This item is identified as a Violation (390/86-20-01), failure to properly change PORC approved procedures.

b. (Open) URI 390/86-18-05; Instrumentation Damage. In inspection report 390/86-18, the inspector identified specific damage conditions and requested verification demonstrating that the supports associated with the valves in question were seismically qualified in the as-installed condition. The inspector noted, that the support for valve 1-FCV-43-2-B was attached to its unistrut support with a strap looped around the operator. The hanger (strap) was loose and allowed the valve to slip out of the strap. It appeared the only support that kept the valve from falling was the instrument line connected to the valve. Valve 1-FCV-43-2-B is attached to the wall as shown in drawing 47A054-42, "Mechanical Category 1 Support Control Air Lines". This is a typical support drawing that has been used for installation of control air lines and valves.

The licensee evaluated the above referenced typical installation for seismic adequacy and on September 12, 1986, reported that the installation does not comply with the seismic qualification requirements for the valve. Also, the installations shown on typical support drawing 47A054-41, "Mechanical Seismic Pipe Support Control Air Lines", fail to comply with the seismic requirements for the valve. The licensee reported these deficiencies on Problem Identification Report number PIRWBNCEB 8684.

This item remains unresolved pending further review by the inspector of the seismic qualification for installed equipment.

4. Unresolved Items

Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve violations or deviations. Two new unresolved items were identified during this inspection and are discussed in paragraphs 5.c and 14.

5. Licensee Action on Inspector Identified Items (92701)

a. (Closed) IFI 390/85-57-01; Lack of overall management of the Employee Response Team (ERT) in that unauthorized maintenance was being performed by the ERT. As stated in Inspection Report 390/85-57, the TVA program for the resolution of employee concerns in the 1985 time frame was very fragmented. In October 1985, TVA established the Employee Concerns Task Group (ECTG) and in February 1986, this program was formalized by the issuance of the Employee Concerns Task Group Program Manual. This manual has been revised several times since its initial issuance in order to make the necessary adjustments for a better delineation of responsibility as well as to modify the mechanism for reporting evaluation results. Although not a part of the described program, normal plant maintenance procedures are utilized by the nuclear power maintenance organization for preparation of areas for reinspection by the ECTG. Additionally, interviews with several task group managers reinforced the fact that any work on plant equipment necessary to support the ECTG is accomplished using established plant procedures.

b. (Open) IFI 390/86-07-01; Followup of new Office of Nuclear Power Employee Concern Program (ECP). The inspector continued to monitor the status of the site employee concern program established in February 1986. As of August 29, 1986, approximately 205 concerns had been identified to the site ECP representative. The site representative indicated that of those concerns identified, 15 had been closed and another 126 are being investigated. The manning of the Watts Bar ECP is currently at four full time investigators and one full time administrative person with one additional investigator reporting at the end of September 1986.

The number of issues being identified and resolved are a concern. The site representative indicated that, of the issues identified, approximately 40% are categorized as Nuclear Safety Related and are receiving priority. Additional temporary help is also planned as a means of reducing the backlog of issues. The inspector will continue to monitor the progress of this program.

c. (Closed) IFI 390/86-14-02; Review of Post Modification Testing Program. The inspectors reviewed completed modification work packages associated with the following safety related modifications:

- Main Steam (system number 001),
- Residual Heat Removal (system number 074).

Selection of modification work packages for review was based on those requiring a functional retest. These packages were evaluated to the criteria established in Administrative Instruction (AI-8.5) revision 19, "Control Of Modification Work On Transferred Systems Before Unit Licensing" and AI-4.1 revision 14, "Quality Assurance Records". Section 5.1.1.d of AI 8.5 states: "...work instructions shall give a step-by-step sequence of events required to perform work correctly, safely and in an efficient manner. They must also ensure that the work is done to meet the applicable specifications, codes, standards, or other requirements; that the new installation is adequately tested; that the environmental qualification of the equipment has not been voided; and that proper quality assurance verification of the work is performed." This requirement is a carryover from prior revisions. Section 5.2 of AI-4.1 requires in part that QA records shall show evidence that an activity was performed in accordance with the applicable requirements, and shall provide adequate information to permit identification between the record and items or activities to which they apply. Also, all blanks shall be filled in or marked 'NA'. Where data is normally required but not taken, explanatory remarks and initials or signature of the person making the remarks shall be entered and dated.

A total of 16 work modification packages were reviewed and 10 appear to be deficient in meeting requirements as referenced in the procedures listed above. The following is a list of those work packages reviewed, deficiencies noted, and comments associated with each:

(1) Unit 1 modification package No. 2914 implemented Engineering Change Notice (ECN) 3179 to provide input to the plant computer alarm anytime that Residual Heat Removal (RHR) flow control valves, 1-FCV-74-8-A and 1-FCV-74-9-B, are not fully closed. The work package failed to reference or include a detailed retest procedure as required by AI-8.5 and AI-4.1.

(2) Unit 1 modification package No. 3431 implemented ECN 3977 to revise control wiring in various valves, including RHR valves, to assure correct remote valve indication. Package documentation does not include an acceptable post test procedure and therefore does not meet the requirements of AI-4.1.

(3) Unit 1 work package No. E-5974-1 provides instructions for relocating steam generator level instrumentation and associated instrument tubing. The package did not include a requirement for reevaluating data necessary to calibrate the relocated instruments as required by AI-8.5.

(4) Unit 1 modification package No. 4186 was implemented to rework conduit which involved terminating wiring associated with 1-FCV-74-24B and 1 FCV-74-12A. This package failed to include a detailed retest procedure as required by AI 8.5.

(5) Unit 1 modification package No. 2209 implemented a wiring change to the Westinghouse Switches (model W-2) associated with various equipment in 10 different systems. Retesting did not meet the requirements specified in AI 8.5.

(6) Unit 1 modification package No. 2035 required electrical cables associated with 1-FCV-74-2B to be removed, conduit reworked, and cables replaced. Retesting specified did not meet the requirements set forth in AI 8.5.

(7) Unit 1 modification package No. 2266 required electrical cables associated with valves 1-FCV-74-15A and 1-FCV-74-16A to be removed, conduit reworked, and electrical cables replaced. Requirements specified in AI 8.5 were not met.

(8) Unit 1 modification package No. 2056 requires the replacement of damaged electrical cables. Affected equipment was not referenced and retest was not specified as required by AI 8.5.

(9) Unit 1 modification package No. 1989 required the rework of instrument lines associated with instruments on various systems. Retest requirements were not specified and requirements of AI 8.5 were not met.

(10) Unit 1 modification package No. 2008 required the removal of electrical cables, conduit reworked and the cables replaced associated with valves 1-FCV-1-103 through 114. Retest requirements were not in accordance with AI 8.5.

The above examples appear to violate stated requirements; however the licensee has indicated that, during the time frame the work was performed, several interrelated processes, included in Nuclear Power and Construction department procedures, were in place. The licensee is currently attempting to assemble all pertinent documentation related to the test packages. Further evaluations will be conducted in this area to determine whether the post modification testing requirements of A.I 8.5 were satisfied by other procedures or documentation. This is identified as unresolved item (390/86-20-02) pending determination by the NRC that adequate testing was performed.

Within this area inspected, no violations or deviations were identified.

6. Followup of Licensee Identified Items (92700)

a. (Closed) CDR390/85-23, 391/85-21; Heating, Ventilating, and Air-Conditioning Duct Insulation Heavier than Weight Used In Support Analysis. This licensee identified item described a condition adverse to quality where the assumed weight of the heating, ventilating, and air-conditioning (HVAC) duct insulation associated with portions of the ducts in the control and reactor buildings was not conservative. The insulation weight and location used in the analysis of typical supports was limited to a maximum of 1.5 lb/ft^2. The licensee determined that in some cases the insulation actually weighed 3 lb/ft^2 in the reactor building and 2.7 lb/ft^2 in the control building. The specified corrective actions included adding nine supports and modifying thirteen other supports associated with the affected duct. The licensee accomplished these modifications in workplans E5827-1 and 8262.

   Workplans, support variance sheets, quality control inspection reports, and checklists associated with completing this activity were reviewed. All areas were found acceptable and this item is closed.

b. (Closed) CDR390/85-24; Deficiencies in Containment Spray Support (CSS) 47A437-1-1. The subject deficiency was initially reported to the NRC in July, 1985. The deficiency involved an as-constructed piping support configuration with overstressed piping support pads and seismically overloaded U-bolts. The licensee evaluated the deficiency and determined that the errors which resulted in the inadequate support, on drawing 47A437-1-1, were isolated cases which occurred several years apart and resulted from human error. Additionally, TVA has reanalyzed, using a new computer code TRIPE, the piping system and determined that the support points located at nodes 2V1 and 4V1 were no longer needed. The support was removed in accordance with ECN 5803.

   The inspector reviewed the licensee's report with regard to the deficiency. The inspector also reviewed drawings 47W437-209 revision 0, "System N3-72-7A Pen X48B Isometric-Static, Thermal & Dynamic Analysis for CSS Heat Exchanger Discharge To CSS Spray Headers" and 47W437-210 revision 0, "System N3-72-8A Pen X48A Isometric-Static, Thermal & Dynamic Analysis for CSS Heat Exchanger Discharge To CSS Spray Headers" and verified that supports were no longer required at locations 2V1 and 4V1 on the containment spray piping. All items reviewed by the inspector were found acceptable and this item is closed.

c. (Closed) CDR390/85-22; Containment Spray Pipe Supports Improperly Mounted. The subject deficiency was initially reported to the NRC in July, 1985. The deficiency involved an as-constructed piping support baseplate incorrectly attached to the shield building rather than to the auxiliary building as required by the piping analysis. The licensee evaluation of this deficiency determined that the incorrectly located support, 72-ICS-R116, was an isolated event attributed to human error by the designer and checker who did not ensure that the design configuration of the support conformed to the piping analysis design. The subject support was relocated in accordance with ECN 5779 as documented by work plan E 5779-1.

The inspector reviewed the licensee's report with regard to the deficiency and also verified that the support had been relocated. All items reviewed by the inspector were found acceptable and this item is closed.

d. (Closed) CDR390/86-01; Incorrect Door Check on Fire Door. The subject deficiency was initially reported to the NRC in November, 1985. The deficiency involved the failure of the door closure mechanism (check) associated with fire door A143 to close the door against airflow resulting from room differential pressure. Further investigation of this deficiency by TVA identified additional closure mechanisms which also required rework, and cases where equipment substitutions had occurred which resulted in installed hardware on fire doors not meeting design drawing requirements. The licensee evaluated the deficiency and determined that the problem was caused by a failure of design to accurately specify equipment requirements and to adequately address the requirement for the closure mechanisms to close against air flow. The subject repairs were performed in accordance with ECNs 5902 and 6150 as documented by work plans E 5902-1 and E 6150-1.

The inspector reviewed the licensee's report with regard to the deficiency and also verified that work had been completed on the door closure mechanism associated with fire door A143. Additionally, the inspector ensured that the associated closure mechanisms were adequate to close against airflow for various fire doors. The inspector noted that the correct hardware had been verified and drawings revised to reflect changes. All items reviewed by the inspector were acceptable and this item is closed.

e. (Open) CDR390/85-46, 391/85-46; Tube Bending Process Deficiencies. The subject deficiency was initially reported to the NRC in September 1985. The deficiency involved instrument line/tube bending operations which did not meet all of the requirements of TVA General Construction Specification G-29, Process Specification 4.M.2.1 requirements. Procedures were in use which did not adequately control the bending process. Specifically, improper bending tools as well as inadequate bender qualification records were used in the field. The licensee evaluated the deficiency and determined that the deficiency was the result of: (1) misinterpretation of G-29 requirements which resulted in inadequate site implementing procedures, (2) failure of personnel to comply with all aspects of existing site procedures, and (3) insufficient training in the use of proper measuring techniques.

The licensee agreed to perform the following corrective action: (1) total reexamination of a representative number (200) worst case instrument line/tube bends to establish that an acceptable level of quality exists in as constructed bends, (2) evaluation by the Office of Engineering (OE) of 21 invalid bending processes to determine acceptability of the processes, (3) conduct additional quality control training, (4) revise various quality control procedures to improve clarity and (5) update associated inspection documentation and process/tool qualification records.

The inspector reviewed the licensee's report with regard to the deficiency. Additionally, the inspector reviewed selected process qualification records as well as the results of several as constructed bend reexaminations. The inspector noted that Quality Control Procedures (QCP) 3.11-2 revision 7 "Instrument Lines", 3.13-5 revision 2 "Instrument Line Bending and Surveillance", and 4.10-5 revision 5 "Pipe Bending" included the necessary revision to more clearly describe requirements. A field inspection of several completed instrument line/tube bends will be performed and this item will remain open pending that followup inspection.

f. (Open) CDR390/86-06, CDR391/86-05; Additional Diesel Generator Relay Not Seismically Qualified. The subject deficiency was initially reported to the NRC in December, 1985. The deficiency involved the use of General Electric (GE) type 12 CFD differential protection relays in the control circuity for the emergency diesel generators. As identified in NRC Information Notice (IN) 85-82, "Diesel Generator Differential Protection Relay Not Seismically Qualified", this type relay is not seismically qualified for class 1E service when in the deenergized mode. Momentary closing of a contact on the relay trips a breaker lockout relay to block automatic operation of the affected diesel generator. The licensee performed an evaluation of the subject deficiency. Based on this evaluation, differences in packaging and location of the relay as used in the licensee's case and qualification documentation provided by the vendor (Northern International, INC.), TVA considers the subject relay installation at Watts Bar as adequately qualified for the intended service and acceptable for use. However, based on conversations with NRC Inspection and Enforcement (I&E) personnel, General Electric Company's position, as stated in General Electric Relay (GER) 3069, is that the GE Type 12 CFD relay is of the fast acting design that is too sensitive. This GE Type 12 CFD relay is not intended for use in this application and should be replaced by a type IJD relay.

The licensee agreed to provide additional documentation to support their position but this was not made available by the close of the reporting period. This item will remain open pending review to determine whether the licensee's evaluation properly addresses the concern as discussed in IN 85-82.

Within the area inspected, no violations or deviations were identified.

7. Fire Prevention and Fire Protection - Unit 2 (42051)

During plant tours, the inspectors observed fire prevention and protection activities in areas containing combustible materials where ignition of these materials could damage safety-related structures, systems or components. The observations included verification that applicable requirements of Administrative Instruction (AI) 9.9 (Torch Cutting, Welding, and Open Flame Work Permit), Standard Practice WB 12.6 (Fire Brigade Instructor's Guide and Fire Brigade Handbook), AI 1.8 (Plant Housekeeping) and WBNP Quality Control Instruction (QCI) 1.36 (Storage and Housekeeping) were being implemented with regards to fire prevention and protection.

During this reporting period, the Inspector reviewed an incident which occurred on August 15, 1986, involving a fire associated with an electrical "short circuit" at the heater junction box on the No. 2 Reactor Coolant Pump (RCP). The fire resulted from an electrical short circuit caused by water from an open piping system entering the electrical junction box when a valve that was tagged closed was opened. The system, Essential Raw Cooling Water (ERCW), had been temporarily transferred from the construction unit to the operations unit by means of the Initial Operation Release Program (IOR).

The inspector's initial review of the event indicated that a breakdown in communications and inadequate procedures were the major contributing factors relating to the incident. A review by the licensee resulted in the issuance of NCR 6966 revision 0 on August 21, 1986. This NCR indicated that a portion of system 67 (ERCW) was transferred under the IOR to assist in cooling the unit 2 Reactor Building. The equipment was not properly tagged with IOR tags by the operations department as required by WBNP-QCI-1.22 revision 8 "Transfer of Permanent Features to the Division of Nuclear Power", and AI 6.5 revision 7 "Procedure for Initial Operation, Testing and Transfer of Equipment and Auxiliaries". The equipment was subsequently tagged and operated by Nuclear Construction (NU CON) per Standard Operating Procedure, (SOP-02) revision 3 "Construction Hold Tags".

The apparent causes of this event were the following: 1) violation of WBN-AI 6.5, 7) violation of WBN-SOP-02, 3) procedural deficiencies in WBN-QCI-1.228, and 4) procedural deficiencies in WBN-QCI-1.60, revision 0, "Work Control". Corrective actions stated on the NCR were: 1) Retrain appropriate Office of Nuclear Power (ONP) personnel on the requirements of WBN-AI 6.5, Section 4.2.4, 2) Retrain appropriate NU CON personnel on the requirements of WBN-SOP-02, 3) Revise WBN-QCI-1.22 to include a more specific description of equipment within the boundaries of operation releases and better verification of tag installation. 4) Revise WBN-QCI-1.60 to emphasize requirements to review workplans for impact on equipment released for operation prior to beginning work. The inspector considers the specified corrective actions appropriate and will monitor implementation during normal plant tours.

Within this area inspected, no violations or deviations were identified.

8. Preoperational Test Program Implementation Verification - Unit 1 (71302)

The inspectors conducted routine tours of the facility to make an independent assessment of equipment conditions, plant conditions, security, and adherence to regulatory requirements. The tours included a general observation of plant areas to determine if fire hazards existed and observation of other activities in progress (e.g., maintenance, preoperational testing, etc.) to determine if they were being conducted in accordance with approved procedures. Also observed were other activities which could damage installed equipment or instrumentation, evaluation of system cleanliness controls and a review of logs maintained by test groups to identify problems that may be appropriate for additional followup.

Within this area inspected, no violations or deviations were identified.

9. Testing of Pipe Supports and Restraint Systems - Unit 1 (70370C)

The inspector toured areas of the Unit 1 auxiliary building and reactor building. Numerous snubbers and restraints were observed. Visual examinations were conducted to check for deterioration and physical damage of mechanical snubbers. Visual examinations were also conducted to check for damage of base support plates, fasteners, locknuts, brackets, and clamps associated with these installed pipe supports.

Within this area inspected, no violations or deviations were identified.

10. Other Safety-Related Piping - Visual Examinations Of Welds (55185 B)

During this inspection period, the inspector held numerous discussions with licensee personnel regarding the reinspection program in the welding area. The reinspections are independently conducted by EG&G through a contract with the Department of Energy. The inspector reviewed the reinspection program in the following areas:

- The reinspection program commenced on April 10, 1986 and on September 15, 1986, of the approximate 1719 components identified for inspection by EG&G, approximately 1549 have been completed. Of those inspected, the licensee's engineering group has performed an engineering evaluation of 638 components and found them acceptable for service. Ten welds identified in Group E, Civil Structures, failed the engineering analysis and are classified "unsuitable for service". The ten welds are associated with the cable tray supports in the control building. On August 29, 1986, Significant Condition Report SCR WBN CEB 8689, "Actual welded connection configuration not installed as described by design drawings and violate allowable stresses per Design Criteria WB-DC-20-9", was issued. Several of these connections proved to be inadequate due to undersize, underlength, and lack of welding.

The inspector selected two of the above welds, identified by EG&G on examination package E0649, and performed a visual examination on the welds. The welds examined are located in the control building, elevation 741, and are support members for cable tray framing. The review found that EG&G had adequately evaluated and documented the deficient condition. Weld number six was underflush by 1/32 inch for the entire length. Weld number three is a fillet weld and not a flare bevel weld as specified on the drawing. In addition, the weld length is less than specified. Welds number three and six were rejected by EG&G on July 10, 1986, for the conditions described above. The EG&G evaluation also identified that these conditions were not identified as deficient on the licensee's quality records.

- The inspector accompanied licensee and EG&G personnel on September 4 and 5, 1986 and witnessed removal of linear indications from welds 1-068A-T129-08F and 1-067B-D209-08. The first weld was a class 1 lug weld located at the top of the pressurizer. The second weld was a class 3 pressure boundary butt weld located in the auxiliary building. EG&G had reported small linear indications in these welds.

  Removal of the first weld indication was accomplished by removing approximately 3/32 inch deep of weld metal. The linear indication appeared on the surface approximately 1/8 inch long and was removed by grinding. After removal, EG&G reinspected the area with a liquid penetrant test and found the weld acceptable. The second weld defect removal observed was on a 16 inch butt weld in the essential raw cooling water line. EG&G reported linear indications located in the weld. The linear indications were removed by grinding in the exterior reinforcement of the weld. After minor grinding, EG&G reinspected the area using a magnetic particle test and found the weld acceptable. All areas observed by the inspector were found acceptable.

- During the week of September 15, 1986, EG&G, through a contract with Hellier Associates, commenced a review of the radiographs for approximately 1700 welds that had been welded and radiographically accepted on-site by the licensee. The decision to perform this review was made when a sample review of eighty six radiographs, by EG&G, revealed three radiographs with indications of unacceptable weld quality. All three welds were originally reviewed and accepted by one licensee's film interpreter. In addition, the same interpreter had accepted all of the 1700 welds discussed above which formed the basis for the 100 percent review. Also, the contractor plans to review an additional 30 radiographs of other radiographic film accepted by different film interpreters to determine if the deficiency is limited to the one interpreter.

Of the areas reviewed, the inspector found that EG&G's weld evaluation program is being performed adequately.

Within this area, no violations or deviations were identified.

11. Reactor Vessel And Internals Work Observation (50053) Unit 2

The inspector observed the reactor pressure vessel, head and internals, to assure proper in-place storage was occurring. Specifically, the following items were observed:

- Ascertain whether protective devices are installed around the top of the open vessel to prevent entry of foreign objects and debris.

- Observe dunnage and support structures for the reactor vessel head and determine whether protection from entry of dirt, water, flooding (height of structure) and strength of support (shifting or collapse of structure) are adequate and consistent with storage specifications.

All areas reviewed by the inspector were found acceptable.

12. Structural Welding - Work Observation (55063C) Unit 2

The inspector selected welds on pipe support number 2-68-402 to perform inspections to ascertain compliance with the licensee's commitments, drawings and associated sketches. The support is located in the containment building elevation 754 feet 1 1/2 inches and shown on sketch number 2-68-402 revision 902, no title. The inspector performed visual inspections of the connecting welds to the existing embedment plate installed in the ceiling for sketch items 5 and 8. The inspector verified that the welds were made in accordance with the specified flare bevel requirements for item 5 and a fillet weld for item 8, and that the welds were made to the lengths specified on the sketch. The support was identified as mark 2-68-401. Further reviews of the documentation found the support was originally installed as 2-68-401 and was later modified and changed to 2-68-402. The inspector found adequate documentation existed to properly identify the support.

Within this area inspected, no violations or deviations were identified.

13. In-depth QA Inspection Of Performance (35061) Unit 1 and 2

During routine inspection activities, the inspector held discussions with Quality Control Inspectors regarding the inaccessibility of inspection procedures to the QC inspector when performing quality inspection functions. The on-site document control of records apparently prohibited removal of a controlled document, such as procedures, from the office for use in the field. This applied to QC inspectors and resulted in inspections being performed without immediate accessibility to the procedure. In some cases the inspectors reprinted certain complex sections, such as tables, from the controlled copy and kept it in their possession for ready reference. In these situations, document control is lost and newly issued revisions might not be noted.

The inspector discussed this concern with the Acting on-site Quality Assurance (QA) Manager and, on September 18, 1986, the Acting QA manager informed the inspector that he would implement a program by September 22, 1986, which allows QC inspectors to take the approved procedure to the inspection point. The inspector was advised that this program change would include the following elements: 1) eliminate inspectors from reprinting copies of procedures that would become uncontrolled, 2) provide an approved copy of the procedure in the field for ready reference to the inspector, and 3) provide a system that is auditable in the field. The implementation of this program will be reviewed in subsequent inspections and is identified as inspector followup item 390/86-20-04 and 391/86-20-02.

14. Verification of As-Builts (37051) Units 1 and 2

The inspector reviewed drawings, field change requests, seismic qualification reports, and field installations of numerous Foxboro transmitters, installed in Units 1 and 2, to determine if the as-installed conditions comply with the seismic qualifications established for these devices.

The seismic qualification requirements are specified in Regulatory Guide 1.100 revision 1, "Seismic Qualification of Electric Equipment For Nuclear Power Plants", which endorses IEEE Std 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations".

The following Wyle Laboratories seismic qualification test reports were provided by the licensee: 1) 42807-1, "Seismic Simulation Test Program on an Instrument Test Rack", for the mounting of Foxboro transmitters and 2) 45592-4, "Qualification Test Report of N-E10 Series Transmitters for Class IE Qualification" for the devices. These qualification reports included attachments such as electrical connections, conduit, and sensing lines to the device which is a requirement of IEEE 344-1975, paragraph 6.1.1, "Mounting", which specifies the equipment to be tested shall be mounted on the vibration generator in a manner that simulates the intended service mounting and the effects of electrical connections, conduit, and sensing lines, etc.,shall be considered.

The inspector noted numerous installations in the Unit 1 containment building of Foxboro Model E 11DM transmitters which were modified by Field Change Request I-1567. This change significantly modified the mounting brackets used in the installation of the devices. The qualification report discussed above (42807-1) did not simulate the observed condition of the transmitters noted by the inspector. As specified in IEEE 344-1975, the mounting during qualification method shall be the same as that recommended for actual service and the mounting method shall use the recommended configuration. The licensee was unable to provide the inspector with any additional information that would demonstrate these installed devices, as modified by the field change request, were still seismically qualified.

The inspector also observed numerous Foxboro transmitters in Units 1 and 2 with electrical connections to the device significantly different from the method used to qualify the devices as specified in Wyle Test Report 45592-5. This report qualified the electrical connection to the device using a 1/2 inch flexible metal conduit and the conduit was attached to the transmitter by means of the conduit connector on each of the conductor seal assemblies. The unattached end of the conduit was permanently affixed to the side of the mounting bracket assembly.

The inspector noted that the Foxboro transmitters were not electrically connected as specified in the Wyle test report. For example, steam generator number 2 level transmitter, 2-LT-3-156, located in the unit 2, north fan room, elevation 719, azimuth 190 degrees, was electrically connected to the device with 15 inches of fitting, solid pipe, and additional conduit. The connection is significantly heavier than the qualified flexible conduit and was not independently supported in the manner qualified.

This item is identified as unresolved item (390/86-20-03 and 391/86-20-01) pending the licensee providing the inspector with the seismic qualification reports and completion of reviews by the inspector.