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

Report Nos.: 50-259/89-17, 50-260/89-17, and 50-296/89-17

Licensee: Tennessee Valley Authority
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1101 Market Street
Chattanooga, TN 37402-2801

Docket Nos.: 50-259, 50-260, and 50-296 License Nos.: DPR-33, DPR-52, and DPR-68

Facility Name: Browns Ferry 1, 2, and 3

Inspection Conducted: May 22-26, 1989

Inspectors: _____ 8/10/89
M. Branch, Inspection Team Leader Date Signed

Team Members: S. Black
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SUMMARY

Scope:

This special announced inspection was conducted in the areas of transitional design change program review, ECN close-out, and review of 10 CFR 50.62 (ATWS Rule) implementation.

Results:

The team inspection concluded that the transitional design control process satisfied the requirements of ANSI 45.2.11-1974 to which the licensee is committed. However, implementation problems associated with circumventing the ECN revision/cancellation process were identified. Additionally, procedural violations associated with the documentation of post-modification testing and with failing to process a field change to reverse electrical leads were identified.

A significant weakness involving 10 CFR 50.59 written safety evaluations was also identified.

Prior to this implementation inspection, TVA's EA group performed an audit of the transitional design change program. This audit identified implementation weaknesses as well as ECN/DCN closure process problems. As part of the licensee's corrective action, the Site Director took a positive step by suspending DNE modification package output until temporary corrective actions and a detailed review plan could be put in place. Implementation adequacy of the licensee's program is unresolved pending evaluation of the licensee's review results and subsequent corrective action.

In the area of the FSAR update process, with the exception of an open issue involving return-to-service closures of ECNs, the licensee's program satisfied the requirements of 10 CFR 50.71. However, the licensee indicated that they had requested an exemption to the annual update of their FSAR pending the validation review to identify and correct FSAR inaccuracies. This exemption will require the review and approval of the NRC licensing group.

Within the areas inspected the following violations were identified:

- Failure to properly implement written procedures as required by Technical Specification 6.8.1 in the areas of DCN/ECN program implementation (paragraph 2.d.), field change requests (paragraph 3.b.), documentation of required post-modification testing (paragraph 3.a.), and intent/non-intent changes (paragraphs 2 and 3)

- Failure to perform written safety evaluations as required by 10 CFR 50.59 (paragraph 2.b.).

- Licensee-identified violation involving inadequate thermal overload calculations (paragraph 2.d)

One unresolved item was identified involving transitional design change program implementation adequacy pending review and evaluation of the licensee's corrective action for the EA audit No. BFT 89901 (paragraph 8).

An inspector followup item was identified involving the FSAR update process (paragraph 6).

A second IFI was identified involving followup on ATWS modifications (paragraph 7).

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*L. Barger, Licensing
*A. Gordon, Acting Manager ISEG
*J. Hutton, Operations Superintendent
*D. Langley, NE EEB
*J. Maddox, NE
*J. McCarthy, Regulatory Compliance Supervisor
*P. Porter, NE EEB
*J. Sparks, System Engineering
*G. Turner, QA Manager
*H. Weber, Engineering/Modifications Restart Manager
*O. Zeringue, Site Director

Contract Employees

J. Isaacs, Bechtel
B. Sharman, Bechtel

Other licensee employees or contractors contacted included licensed reactor operators, auxiliary operators, craftsman, technicians, and quality assurance, design, and engineering personnel.

NRC Resident Inspectors

B. Bearden
K. Ivey
*C. Patterson

*Attended Exit Interview

Acronyms used throughout this report are listed in the last paragraph.

2. Design Change Process Review (37700)

a. Process Evaluation

Using a Design Change Process flow chart developed by TVA, the inspector reviewed the process depicted versus the transitional design change process described in the NPP, Volume III, Section 2.3.1. The scope of the review was limited to those activities performed by NE during the preparation, review, and approval of ECN modification packages and DCNs. The inspector's

review include a detailed evaluation of the following upper-tier and lower-tier design engineering procedures:

NEP 3-1, "Calculations", Revision 1-PCN-4

NEP 5-2, "Review", Revision 0-PCN-1

NEP 6.2, "Design Change Notice", Revision 0-PCN-3

NEP 6.3, "Operating Plant Modifications", Revision 0-PCN-2

NEP 6.6, "10 CFR 50.59, Safety Evaluations", Revision 1

PI 86-03, "Preparation and Control of Engineering Change Notice ECN Modification Package", Revision 7

PI 87-41, "Design Change Notice", Revision 3

PI 87-54, "Performance Task Contractor Manual", Revision 2

The transitional design-control system was based on modifying the existing TVA design-control system; facilitating a transition to the permanent TVA system; and providing comprehensive design change packages. Based on review of the above procedures, the inspector determined that design changes to the plant could be made under this system by any of the following modification processes:

ECNs
H-DCN
W-DCNs
F-DCNs

The controlling procedure for the preparation and approval of ECNs was PI 86-03. This procedure established the design-change controls necessary to ensure that the BFN design baseline and as-constructed configuration are maintained during the design process. Responsibilities of persons involved in the design-engineering process were identified. Additionally, the scope of the activities to which the design-engineering controls are applicable was specified. Paragraph 4.1 established requirements that ensure applicable design-inputs are identified, documented, and their selection reviewed and approved. This is accomplished by completion of Attachment C, Modification Criteria. Completion of additional attachments, e.g. Attachments M, N, O, and P, ensures that the design analysis is conducted in a planned and controlled manner. Provisions for performing a screening review, and 10 CFR 50.59 Safety Evaluation if required, were specified in paragraph 4.1.9. Independent Design Verification was performed in accordance with the requirements of NEP 5.2. and paragraph 4.1.14.

Based on review of procedure PI 86-03, no design control program deficiencies were identified.

The preparation, review, and approval of DCNs is the process by which changes are made to ECNs. Procedure PI 86-03, paragraph 4.2, addressed these controls. The controlling procedure for the preparation, review, and approval of DCNs was PI 87-41. This procedure provided project-specific clarifications, responsibilities and supplemental requirements necessary to implement the DCN process specified in NEP 6.2, and the referenced paragraphs of NEP 6.3. The scope of the activities to which the design controls are applicable is specified in paragraph 2.0 of the PI. This paragraph stated in part that a DCN which causes a plant modification must be authorized by a DCR, FCR, or a plant initiated DCN (H-DCN).

DCNs were classified as either W-DCNs, H-DCNs, or F-DCN. The definition of the various types was contained in paragraph 4.0 of PI 87-41. This definition is not consistent with that contained in NEP 6.2, paragraph 2.1. However, the controls specified in PI 87-41 were applicable to the processing of F-DCNs. DCNs initiated by NE (W-DCNs) or the plant (H-DCNs) were processed in accordance with the design controls specified in NEP 6.8, paragraph 7.c. This procedure required that minor plant modifications, including changes to design documentation, shall be made via the DCN process. The minor modification criteria specified on Attachment 7 must be met for proposed changes dispositioned by this process. Additional design controls were established to ensure performing a screening review and completion of Attachment C, "Modification Criteria".

Based on review of the above program documents, no design control program deficiencies were identified.

b. 10 CFR 50.59 Safety Evaluation Review

The inspector evaluated the USQD process used by Design and verified that it met the requirements of 10 CFR 50.59.

The requirements to perform the reviews specified in 10 CFR 50.59 were discussed in the licensee procedure NEP 6.6, "10 CFR 50.59 Safety Evaluations". The TVA program required two reviews. The first was a screening process which required only the proposing organization to determine if the proposed change was within the scope of 10 CFR 50.59. The second review only occurred if the first review was positive and was an evaluation to determine if the proposed change contained a USQ. This evaluation was performed by the proposing group, received a cross-disciplinary review and plant manager approval, and was reported to the NRC. This two part program appeared to have been established to eliminate the need for the more involved, higher level evaluation of simple facility changes that are not described in the FSAR.

The NRC inspector reviewed four DCNs to verify proper procedural implementation of NEP 6.6 requirements. These were:

DCN H 3858A	-	This change added a 0.75 second time delay to the auto start logic of HPCI and RCIC.
DCN P 7113	-	This change added air dryers to the diesel generator starting air system
DCN H 0166A	-	This change added relief valves on the discharge of the drywell air compressor
DCN H 1654A	-	This change modified the failure mode of the water supply valves of the diesel generator building ventilation system chillers.

The NRC inspector determined that three of the four screening reviews performed for the above listed changes did not meet the requirements of NEP 6.6, in that the screening reviews failed to require that a safety evaluation or USQD be performed for those facility changes that were described in the SAR. DCNs H 3858A, P 7113, and H 0116A were either described in text or drawings of the SAR.

Additionally, ECN P-7067 was written and implemented to add a single line of sprinklers for coverage of the core spray valve area in the reactor building. The screening review form, B22880517511, determined that a safety evaluation was not required for this modification. The screening review asked the following question:

> Does the proposed change involve a change in the facility (or plant operating characteristics) from that described in the SAR which could impact nuclear safety?

The justification for the negative answer to the question included the following statement:

> The effects of water spray on safety-related equipment following a failure or actuation of the seismically supported fire protection piping system will be evaluated by TVA under Contract Number TV-73039A (See letter from P. J. Speidel to R. E. Gallagher dated March 15, 1988 -- RIMS No. B22 880315 020)...

This statement implied that the effects of the installation of this piping had not been fully analyzed, but the fact that a contract existed and the effects would be analyzed was given as justification for not performing a safety-evaluation on this modification. Since the effects were unknown at the time of the screening, the correct answer should have been positive, stating that the change could impact nuclear safety, in order to be conservative.

In discussions with TVA regarding this issue, the fact was revealed that a significant revision to NEP 6.6 had become effective in April 1989. This revision was intended to clarify the screening process to ensure that items such as those identified above would be captured for the USQD process. The screenings of the DCNs reviewed were performed in mid year 1988. There was insufficient review material provided in this inspection period to review and evaluate post revision screenings. The failure to properly implement, for the examples above, the requirements of procedure NEP 6.6, which required written safety evaluations for the modifications performed, is identified as violation 50-259, 50-260, 50-296/89-17-01, Failure to Comply With the Requirement of 10 CFR 50.59.

c. System Design Criteria Document Review

For the Standby Liquid Control System, System 63, the inspector reviewed the DCD. This review was to determine if information required by Section 3.2 of ANSI 45.2.11 - 1974 was included.

The TVA requirements for all design criteria documents at operating nuclear plants had been provided in NEP 3.2, "Design Input." That procedure contained requirements to include or justify excluding all of the attributes discussed in Attachment 1 of the procedure. Attachment 1 contained all of the items required by Section 3.2 of ANSI 45.2.11 - 1974, "Quality Assurance Requirements For The Design of Nuclear Power Plants."

The restart design criteria documents were established to meet the commitments of the DBVP described in Sections 2.2.1 and 2.2.2.1 of Volume III of the NPP. That program was established to eliminate weaknesses that existed in previous design criteria documents, such as lack of a design basis to evaluate new design changes, unimplemented design changes, and field changes; and a lack of a consistent and comprehensive information system to manage the design data base.

These weaknesses were attributable to root causes such as "...a lack of detailed design output, and the absence of a centralized design basis." Furthermore, the design criteria and design basis information had not been kept up-to-date and were difficult to utilize.

The purpose of the DBVP was to reconcile engineering design documents, including supporting essential calculations, design criteria, and licensing requirements in such a way as to eliminate the existing program weaknesses. Procedures were developed to define the licensing commitments and technical-requirement review process and control preparation of design basis documents. Design basis documents include: system design criteria documents; general design criteria documents; system requirements calculations; control room drawings including flow, control, and single line drawings; and a list of essential calculations.

The design criteria document for the Standby Liquid Control System, System 63, was BFN-50-7063. The NRC inspector found that the restart design criteria acted as a focal point for design commitments and requirements and as such contained very few details or system specifics. For example, NEP 3.2 required system material requirements to be specified, including such items as compatibility and corrosion resistance. Section 3.12 of BFN-50-7063 specified material requirements as follows:

> Specific material requirements for components of the SLC system are identified by the original bills of material, vendor documentation, specifications as they exist, and original purchasing requirements, plus additional material requirements covered by Commitments/Requirements made to later editions of codes, Standards, and Regulatory Guides.

Various comments were made through the design criteria document regarding materials; however, all referred back to Section 3.12 for details. The format of the document was not conducive to easy review.

In general, however, the NRC inspector found that the restart design criteria document BFN-50-7063 and its references did address the design input requirements of ANSI 45.2.11 - 1974 as stated in NEP 6.6.

No violations were identified.

d. ECN/DCN End Product Review

The inspector selected ECN P 7010, and DCN H 1239, Revision A, for review. This review was to evaluate the end product against the process and included the following:

- Verification that the establishment process ensures that original design information was available to the design change group. Also, verification through interviews, for a sample of contractors performing design work, that access to original design information was readily available.

- Evaluation of the controls of the Design Analysis to ensure that:

  - They were performed in a controlled and planned manner.

  - Design analyses were controlled as QA records. (Several were sampled to ensure they are legible, suitable for reproduction, retrievable, and technically adequate.)

- The Program required documentation of analyses to contain:

  Method of analysis
  Purpose
  Assumptions
  Basis or design input
  Person performing analysis
  Date
  Reviewer
  Results or conclusions

° Evaluation of controls to ensure that equipment accessibility for maintenance, inservice inspections, and replacement if necessary was considered in the design process.

° Review of the design verification process and ensure that it required design verification by independent design review, alternate calculations, or qualification testing.

1) ECN Number E-2-P 7010, Revision 0

The above ECN was prepared to provide design basis documents that showed design-verified thermal overload heater size and setting for motor control centers required to support Unit 2 restart. Responsibility for preparing the ECN was assigned to a licensee contractor in accordance with Task Scoping Document TSD-E034, "Thermal Overload Heater Documentation", dated July 13, 1987. The detailed task description specified activities to be performed by the contractor and included the preparation of TOL calculations. Subsequent to the completion of the design-engineering activities for ECN P 7010, licensee management identified an error in the calculation used for determining the TOL relay size and trip current setting. CAQR No. BFP 850447, dated June 27, 1988, was prepared by the licensee to document the design deficiency and initiate corrective action.

The root cause of the design deficiency was identified as improper use of General Electric Heater Tables. The error involved the calculation of the TOL relay trip setting as 1.25 times the maximum motor full load current listed in the heater tables. The correct value is calculated as 1.25 times the heater minimum current. The inspector reviewed selected copies of the calculation and verified that the calculations had been revised to incorporate guidance from the vendor contained in a General Electric Application Tips letter dated March 11, 1988. Corrective action for this design deficiency was completed with the issue of the revised calculations. This design deficiency was characterized as a Licensee-Identified Violation, 50-259, 50-260, 50-296/89-17-03, Inadequate Thermal Overload

Calculations. This violation met the criteria specified in Section V of the NRC Enforcement Policy for not issuing a Notice of Violation and was not cited.

2) DCN H1239, Revision A

The above DCN modification package was prepared to revise the design output drawings showing design verified TOL relay sizes, trip current settings, and bill of material. This DCN superseded all DCAs contained in ECN modification package E-2-P 7010. Pursuant to discussions with licensee management and review of selected samples of DCAs contained in the DCN, the inspector verified that the design scope of DCN H 1239 was identical to that of ECN E-2-P 7010. Also, the DCN drawings showed numerous changes in heater sizes and settings from those shown on drawings contained in ECN P 7010.

The transitional design controls under which DCN H 1239 was prepared and evaluated were reviewed by the inspector. Procedure PI 86-03, paragraph 4.2, "Processing Changes to ECNs," did not permit the use of a DCN to correct ECNs for which the initial screening and/or 10 CFR 50.59, Safety Evaluation was no longer valid. Because the ECN P 7010, 10 CFR 50.59 screening review and Safety Evaluation were based on inputs from TOL calculations that contain errors, the inspector concluded that the result of this review was incorrect.

Discussions with licensee management concerning the reason why a DCN was prepared in lieu of revising ECN P 7010 were conducted. The inspector determined that the DCN process was used because it was the most expeditious way to implement field changes required to support Unit 2 fuel load. This failure to comply with the transitional design controls was identified as the first example of Violation 50-259,50-260,50-296/89-17-02, Failure to Properly Implement Procedures As Required By TS.

e. Material Selection Review

The inspector reviewed the process used to specify and procure materials for an ECN. The inspector reviewed the parts that the design engineer had specified for ECN P-7032 and the parts that had actually been used for the ECN. The design engineer had specified the material to be used as required by SDSP-16.2, "Procurement of Material, Components, Spare Parts, and Services", Revision 0. The inspector reviewed the QA levels specified by the engineer for the different components and found them in line with the safety requirements of the components and systems involved. All the components drawn from Power Stores met or exceeded the requirements of the design engineer. No problems with the specification and use of materials were identified.

f. Interface Control Review

The inspector reviewed a sampling of recent modifications to evaluate engineering discipline interface controls. These modifications involved ECN P-7032, which dealt with the upgrading of certain Reactor Water Cleanup System cables to meet 10 CFR 50.49 requirements. During the pre-implementation review of the modification, it was determined that there would be four modifications being performed in the area at the same time. These modifications were being developed and coordinated by different design groups. Even though it was recognized before implementation that interference would exist due to the other modifications, work proceeded. These interferences created by the other modifications resulted in numerous field revisions to the ECN, including the rerouting of the cable conduit, splicing the repulled cables when they were too short to follow the rerouted conduit, and multiple cases where conduit supports would have to be relocated. The lack of coordination among the various design groups resulted in the task becoming more complicated and requiring many field revisions. The licensee has since terminated the use of several of the various design organizations, a move which has the potential for reducing the number of interferences and required field revisions on the modification process.

The inspector also reviewed the process used to develop workplans for approved modifications. Licensee procedure SDSP-8.2, "Modification Workplans", Revision 13, included an attachment which provided general requirements for all workplans. All safety-related workplans were required to have a review by a technical reviewer, the post-modification test manager, and the site quality organization.

Changes to a workplan were implemented as either an intent change or a non-intent change. An intent change was defined on Form SDSP-122 as:

Removal of an item installed by some other work document.

Change to acceptance criteria.

Deletion or change to a QC or ANI holdpoint. This may be processed as a non-intent change if QC/ANI preapproval is obtained.

Changes in scope technique or sequential order of instruction steps that would affect the results or nuclear safety.

Changes which would implement a temporary alteration to a CSSC without a TACF.

Changes to the authority or responsibility for review and/or approval of the document, or the results obtained from its implementation.

This definition of intent change provided a great deal of ambiguity for the classification of a change to a workplan. The inspector found several examples where significant changes were made to workplans and were classified as non-intent changes. These changes did not receive the level of review and approval necessary for the original workplan. Examples of these significant changes included change number two to workplan 2317-88, which deleted a support drawing and weld map from the original workplan, multiple changes to workplan 2069-88 to allow such things as splicing of short cables, abandoning instead of removal of cables, making as-needed repairs to concrete, and cutting out an existing weld and rewelding. Another example concerning the switching of leads at a breaker panel is discussed in paragraph 3 below. The failure to properly change work plans is identified as an additional example of violation 50-259, 50-260, 50-296/89-17-02, Failure to Properly Implement Procedures as Required by TS.

3. Post Modification Testing

The inspector selected nine recent modifications for review which required some type of post modification test. Each was reviewed to determine the adequacy of testing to insure that the affected area had been properly tested and met design requirements.

The nine modification packages reviewed by the inspector are listed below:

Work Plan	ECN/DCN	Description
WP 2600-88	ECN P 7131	Reroute the unit 2 reactor vessel level reference piping from the vessel penetration to the first isolation valve outside the drywell penetration.
WP 0132-88	DCN W 0113A	Replace GE capacitors inside the 250 Volt D.C. battery charger, 2A, with equivalent Mepco or GE capacitors.
WP 2181-88	DCN WW 044A	Modify CSSC motor operated valves control switch settings based on valve vendor data and criteria established to permit valve seating/unseating without exceeding the rating of the valves and to prevent inadvertent backseating.
WP 2010-88	DCN B 0013C	Rework thermowells in the RBCCW system and four temperature elements in system 68.
WP 2134-88	ECN 7013	Replace internal wiring in Limitorque Operators.

WP 2194-88	DCN-P 7082, R-6	Replace existing Reactor Water Clean-Up pump motors, U-2.
WP2323-88	DCN WW 0186AA	Replace the starter coils, add an interposing relay in compartment 10C, and replace the starter coil in compartment 11A of the 480V Diesel Auxiliary Board B.
WP 2340-88	DCN W 0557A	Provide control switch settings based on vendor data: criteria established by G-50 on motor operated valves.
WP 1100-88	DCN H 1238A	Incorporated design verified overload heater sizes for motor control centers.

Of the nine packages reviewed, the inspector determined that the work plans contained the following deficiencies:

- WP 2181-88, which had changed valve limit and torque settings, had not specified a leak rate test for valve 2-FCV-71-34. As part of the field completion package, Site Directors Standard Practice 8.4, Revision 13, "Modification Workplans", required the responsible engineer to assure that all documentation is complete. This assurance was made without all required post modification testing being specified or completed. This failure to follow procedure is an additional example of violation 50-259, 50-260, 50-296/89-17-02, Failure to Properly Implement Procedures as Required by TS.

- WP2194-88 incorporated the design necessary to change out the Unit 2 reactor water clean-up pump motors. During the inspector's review of the post modification testing, it was learned that after the motor changeout was completed, checked for rotation, and the electrical leads spliced, it later became necessary to reverse the electrical leads. This time the leads were reversed by changing them at the breaker and this was accomplished by adding a step in the work instructions. After the leads were reversed, drawing 67 E 2-45N2748-4 was not changed to reflect the actual installation which required the color coded leads to be terminated at specific breaker terminals.

  Section 6.5 of SDSP-8.4 states that if problems are encountered during implementation of the workplan the responsible engineer shall determine if a design field change is required. The reversing of the leads without a field change being processed resulted in drawing 67E 295 N 2748-4 not being corrected. This is an additional example of Violation 50-259, 50-260, 50-296/89-17-02.

4. Drawing Update Process

The NRC inspector reviewed the drawing/procedure update process to verify that adequate controls were in place to ensure changes to the plant are incorporated into the drawings and procedures prior to declaring systems operable.. The inspector sampled six modifications and verified that control room drawings and plant procedures were changed prior to declaring the systems operable. The packages reviewed included:

REN P 7065 DCN H 0130A
REN P 7131
REN P 7045
REN P 3098
REN P 7044

The inspector selected ten systems to verify that a complete set of Control Room drawings existed, including flow, logic, schematic, and single line electrical, and to verify that they were clear, legible, and reflected the latest modification to the system. The systems reviewed were:

Sys. 82, D G Fuel Oil
Sys. 67, Emergency Equipment Cooling Water
Sys. 63, Standby Liquid Control
Sys. 30, Heating and Ventilation
Sys. 65, Standby Gas Treatment System
Sys. 85, Control Rod Drive
Sys. 74, RHR
Sys. 75, Core Spray

The inspector reviewed approximately 50 separate primary drawings in the control room for legibility and inclusion of the latest modifications. This review included flow diagrams, instrument logics, and electric schematics. All drawings reviewed were clear, legible and were updated within the required time frame to include the most recent modifications. Revision clouds were limited to the most recent modifications. Drawing deviations still under review by engineering were clearly marked. The extensive backlog of drawing deviations still under review could compromise drawing accuracy. Several instances of deviations as old as 1985 were noted as still requiring evaluation. The impact of these deviations were minor in each case and none had direct operability implications.

The inspector reviewed the drawing and procedure update process to ensure adequate controls were in place for incorporating changes prior to declaring systems operable after modifications are complete. Operability checklists are used to ensure that modifications are reflected properly on drawings prior to declaring systems operable. The process involved the Systems Engineering group as coordinators of the checklist packages. Reviews of checklist packages did not reveal instances of incomplete or inaccurate operability determinations. This process appeared to be adequate.

The Mechanical Logics series had been removed from the Primary Drawing list and were not being updated. This decision was based on a determination that the Logics had not been properly updated in the past and had consequently been allowed to become obsolete and inaccurate due to modifications performed since licensing. Plant management was considering whether the Logics should be restored and, if so, the appropriate completion schedule. The inspector questioned several members of the Operations staff concerning the removal of the Mechanical Logic (47E-611 series) Diagrams. The use of the Logic diagrams at Brown's Ferry by the Operations staff was not extensive. In fact, several operators did not realize that the Logics had been removed from the Primary Drawing file. Operations personnel were not trained on the Logics, but used flow diagrams, schematics and instrumentation logics instead. Therefore, the removal of Mechanical Logics did not appear to be an issue from the perspective of the operators. The impact of removing Mechanic Logics on other groups such as Design and Modifications personnel should be evaluated as part of the determination of whether the Mechanical Logics will be restored.

The simulator update process appeared to be adequate. Some simulator modifications were being delayed, and apparently this delay was caused in part by recent staff reductions.

The large backlog of approximately 1500 restart-required drawing deviations and the low work-off rate may have an impact on the planned startup date.

5. ECN/DCN Closeout Review

This portion of the inspection was to assess the licensee activities associated with closeout of ECNs. A brief description of identified problems associated with ECN/DCN closeout along with the licensee's closeout process is discussed below.

The ECN/DCN is the vehicle used by TVA Engineering to evaluate and approve changes to the physical plant. Changes to the facility are allowed by 10 CFR part 50.59 provided the written safety evaluation for the proposed modification determines that an unreviewed safety condition will not be created by the modification. Additionally, 10 CFR Part 50, Appendix B, Criterion III, Design Control, requires that design changes, including field changes, be subject to design control measures commensurate with those applied to the original design. To implement this design change into a plant modification, additional programs and procedures are utilized.

The modification program converts the design change into a physical plant modification. However, in the past, procedures necessary to feedback to the design organization the actual modification versus the proposed modifications were ineffective. This resulted in poor communication between the design and operating organizations in the area of design control. An essential part of this operations/design feed back process should be the closure of the ECN which then establishes the design basis for additional modifications. Currently, only 746 of the 1758 Cycle 5

ECN/DCNs have been closed. TVA developed closeout procedure PI 88-04, which engineering follows to review and close ECN/DCNs. In the past this closeout review was only a paper review to ensure all elements associated with the ECN/DCN process were complete. However, due to a recent EA audit this procedure was being revised to require a technical review as well as the currently required administrative review.

The licensee stated that all Cycle 5 restart ECN/DCNs will be closed prior to restart. If new ECN/DCNs are opened as a result of the partial closure of an existing ECN/DCN, they will be evaluated against the restart criteria and scheduled accordingly. If an ECN/DCN requires partial closure, the current TVA practice is to redefine the scope to cover only the completed work, revise the USQD evaluation, close the ECN/DCN and open another one to cover any unimplemented work. The inspector found the commitment to completely close all Cycle 5 restart ECN/DCNs prior to restart acceptable. The inspector could not verify that TVA was making adequate progress toward closing the backlog because there was no schedule available at the time of the inspection.

6. FSAR Update

10 CFR 50.71 requires that an annual update be submitted to the FSAR. The inspector reviewed the licensee's design and licensing controls to ensure the requirements of 10 CFR 50.71 were being properly implemented. TVA used procedures SDSP 15.7 and NEP 3.2 to update the FSAR. Revisions to the FSAR text and drawings were prepared as part of the original ECN/DCN. However, the revisions were held and not incorporated into the FSAR update until the associated ECN/DCN were closed by DNE. Currently, DNE does not review the proposed FSAR changes for accuracy when it closes ECN/DCNs, only that the update request has been made.

The inspector found that there were currently 92 ECN/DCNs which were classified RTS but not closed because of unverified assumptions, and 65 safety PMENs which had been implemented on a safety train basis but the ECN/DCN remained open until all work was completed. These ECN/DCNs could potentially result in a plant configuration which is not reflected in the FSAR, should these packages result in a RTS or PMEN prior to January 22, 1989 (the cut-off date for the UFSAR process) whose ECN closure occurs after that date. The procedures required ECN closure before a UFSAR update is processed. TVA should ensure that the updates to the FSAR are incorporated based upon return to service of equipment instead of ECN/DCN closure which could lag behind. This is identified as IFI 50-259,260,296/89-17-04, Changes That Require FSAR Update.

TVA requested a one year schedule exemption from the requirement to update the FSAR by July 22, 1989. The inspector discussed the plans for the FSAR validation and update; however, no formal documentation existed for the inspector to review.

7. Inspection to Determining Compliance With ATWS Rule, 10 CFR 50.62 (TI 2500/20: Revision 1).

In order to evaluate the licensee's implementation of the ATWS rule, the inspection was conducted using inspection guidance contained in NRR Temporary Instruction 2500/20, Revision 1 and included review of the licensee approved design, verification that modifications did not compromise the safety features of existing safety-related protection system, verification that commitments made to satisfy SER requirements were implemented, and verification that the ATWS system was designed, procured, installed and tested under an approved QA program that satisfies the requirements of GL 85-06.

a. Summary of Implementation Status

1) The Standby Liquid Control System was installed and was declared operational. The Technical Specification had been approved by NRC. The operators were trained on the Standby Liquid Control System emergency operating instruction EOI-1 and 2-OI-63. Surveillance Instructions SI-4.4.A.1, A2, C2, C3, C4, & D had been issued.

2) The ARI installation was not complete and the system had not been declared operable.

a) ARI vent valves and instrument air tubing were installed.

b) Local racks were installed, electrical conduits were installed, and transmitter and trip units were installed. However, most connecting cables had not been pulled into the conduit. Relays and terminal blocks had not been installed in local racks.

c) Control room manual initiation switches, indicators, and alarm window were not installed. However, the modification had been reflected on the simulator control panel.

d) ARI function had not been implemented in a procedure or training document.

3) The RPT installation was not complete and the system had not been declared operable.

a) The recirculation pump trip related to "end of cycle" trip function was available either automatically or manually from the control board.

b) Modification related to ATWS rule requirements had not been completed. It was in the same status as the ARI system.

b. Plant Specific Design Requirement Inspection

The plant specific design areas reviewed and the inspection results are discussed below:

1) Objective

Examine vendor documentation to verify that adequate hardware/component diversity exists between the ARI/RPT equipment and the existing reactor protection system equipment.

Physically inspect the ARI/RPT system and RPS cabinet equipment to further confirm that the required hardware/component diversity exists.

Results

Through review of the documentation in the modification package, the inspector determined that the ARI/RPT systems used Agastat GP series relays. The same type of relays were also used for the RPS Analog trip unit actuation circuits. The ARI vent valve used ASCO solenoid valves (Part No. ASCO THC8316E36, THC8210878). The same type of ASCO valves were used for the RPS backup scram vent valve. The Rosemount ATTU were used for both the RPS and the ARI/RPT system. It appeared that some components in the ARI/RPT system were not diverse from the RPS component. This item is open.

2) Objective

Review support documentation (electrical schematics, power distribution drawings, etc.) to confirm that the logic power supplies selected for the ARI/RPT logic circuits provide the required independence/separation from that associated with the RPS in accordance with the ATWS requirements.

Physically inspect the ARI/RPT logic power supply wiring, including power cables entering the associated cabinets to verify the independent/separate power source to the ARI/RPT system.

Results

Through review of the documentation of the modification package, the inspector determined that the RPS logic and instruments were powered from the RPS MG set AC power panel, while the ARI/RPT system logic and instruments were powered from the Class 1E 250V DC panel. The design satisfied the power independence requirement.

3) Objective

Physically inspect and trace as appropriate the ARI/RPT input and output wiring to verify the use of the reactor water level and the reactor pressure instrumentation for input signals and the use of ARI/RPT output signals to actuate the ARI valves and the RPT breaker circuits.

Examine electrical schematics to confirm this portion of the design.

Results

The RPS and the ARI/RPT system used separate sensors. The logic circuit for the RPS and the logic for the ARI/RPT system were located in separate cabinets. All the RPS cables were in conduits. The ARI/RPT circuits were in separate conduits. The ARI/RPT system was physically separated from the RPS.

4) Objective

Verify that the existing RPS separation criteria continues to be met subsequent to the implementation of the ARI/RPT equipment. This should include a description of the existing plant separation criteria and an inspection of the ARI/RPT wiring to verify consistency.

Results

The inspector reviewed the existing plant separation criteria and the ARI/RPT modification package. It appeared the modification for the ARI/RPT system will not violate the separation criteria.

5) Objective

Review how maintenance will be performed with the reactor at power for the ARI/RPT system. Confirm that maintenance procedures are in place. Inspection the hardware (control switches, alarms, indication) in place which will allow for the performance of maintenance while at power. Verify that the hardware implemented for maintenance is consistent with the human-factors guidelines in effect at the plant. Inspect the bypass controls and verify that no jumpers or lifting lead methods are being utilized for bypassing.

Results

The ARI/RPT modification had not been completed. No procedure was available for review. This issue is open.

6) Objective

Inspect the existing manual control room controls associated with the ARI/RPT system. Review the emergency procedures for an ATWS event.

Results

The hardware had not been installed. No emergency procedure was available for an ATWS related event. This issue is open.

7) Objective

Verify that preoperational testing was accomplished for the ARI/RPT prior to plant startup subsequent to the ARI/RPT system implementation. Review the planned at-power testing procedures, and the hardware design (with supporting schematics) with respect to the capability to test ARI/RPT system during both power operation and while the plant is shut down. Examine the restriction for allowed out-of-service times, during testing and for an inoperable ARI/RPT. Inspect the hardware permanently installed as part of ARI/RPT to accomplish testing including control room annunciation, indication, controls, and bypasses. Review the administrative test procedures to verify that ARI/RPT will be returned to service upon test completion. Observe representative test.

Results

The ARI/RPT system had not performed preoperational test. This issue is open.

8) Objective

Review the circuitry schematics which allows for the completion of mitigating action once the ARI/RPT function is actuated. Review electrical schematics as necessary to verify this phase of the design. Examine the required deliberate operator action that must take place to return the final actuation devices to normal status upon completion of the required action.

Results

Once the ARI/RPT system initiated, it will go to completion of mitigative action. Deliberate operator action must take place to return the final actuation devices to normal status.

9) Objective

Verify that appropriate operating and maintenance procedures are in place and that personnel had been sufficiently trained to assure satisfactory performance of the installed ARI/RPT system.

Results

No procedures were available at this time. No training was provided to the operator on the ARI/RPT system. This issue is open.

10) Objective

Verify that the actual ARI function time test was performed, and the test results met the design requirement that the control rod motion will begin less than 15 seconds after ARI initiation and will be completed within 25 seconds.

Results

The ARI function time testing had not been performed. This issue is open.

11) Objective

Examine documentation (plant procedures, calculation data sheets) to verify sufficient Boron to achieve Hot Shutdown. Verify that operators are trained to apply standby liquid control system.

Results

The SLCS has been declared operational. The Boron concentration calculation method was verified. The operators had been trained to apply SLCS by emergency operating instructions EOI-1 and 2-OI-63. Surveillance Instructions had been issued for quarterly system functional test and 18 months operating cycle functional test.

c. Conclusion

Since the electrical installation had not been completed for the ARI/RPT system, this inspection did not accomplish the original objective. A follow-up inspection is required after the licensee declares the ARI/RPT system operational. This is identified as IFI 50-259, 50-260, 50-296/89-17-05, Followup on ATWS Modifications. The following items need re-inspection:

1) Component Diversity*
5) Maintenance Procedure
6) Manual Control in the Control Room
7) Preoperational Test
9) Procedure/Training
10) ARI Function Time Test

*A Generic Letter is being prepared by the NRR to require the licensee to certify that the ARI/RPT system satisfies the diversity requirements.

8. Review of EA Audit BFT 89901 - Design Change Control

Prior to this NRC inspection, TVA's Engineering Assurance group performed a technical audit of the transitional design change process. This audit resulted in seven CAQRs which documented specific deficiencies in the implementation of the transitional program. As a result of these deficiencies the Browns Ferry Site Director suspended the issuance of modification packages until temporary corrective action and a review plan to determine extent of the condition could be developed. The inspector will review the licensee's corrective action in a future inspection. This item is identified as unresolved item 259,260,296/89-17-06, Followup of Licensee's Corrective Action for EA Audit, BFT 89901.

9. Exit Interview

The inspector scope and findings were summarized on May 26, 1989, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection results listed below. The licensee did not identify as proprietary any of the material provided to or reviewed by the inspectors during this inspection. Dissenting comments were not received from the licensee.

Item Number	Description and Reference
259,260,296/89-17-01	Violation - Failure to Comply With the Requirements 10 CFR 50.59, Paragraph 2.b
259,260,296/89-17-02	Violation - Failure to Properly Implement Procedures as Required by TS, Paragraphs 2 and 3
259,260,296/89-17-03	Licensee identified violation - Inadequate Thermal Overload Calculations, Paragraph 2.d
259,260,296/89-17-04	IFI - Changes That Require FSAR Update, Paragraph 6
259,260,296/89-17-05	IFI - Followup on ATWS Modifications, Paragraph 7
259,260,296/89-17-06	URI - Followup of Licensee's Corrective Action for EA Audit, BFT 89901, Paragraph 8

10. Acronyms and Initialisms

ANI - Authorized Nuclear Inspector
ANSI - American National Standards Institute
ARI - Alternate Rod Injection
ATTV - Analog Transmitter Trip Unit

ATWS	-	Anticipated Transient Without Scram
BFN	-	Browns Ferry Nuclear
CAQR	-	Condition Adverse to Quality Report
CFR	-	Code of Federal Regulations
DCA	-	Design Change Authorization
DCD	-	Design Change Document
DCN	-	Design Change Notice
DCR	-	Design Change Request
DNE	-	Division Nuclear Engineering
DBVP	-	Design Baseline Verification Program
EA	-	Engineering Assurance
ECN	-	Engineering Change Notice
FCR	-	Field Change Request
F-DCN	-	Field Initiated - Design Change Notice
FSAR	-	Final Safety Analysis Report
GE	-	General Electric
H-DCN	-	Plant Initiated Design Change Notice
HPCI	-	High Pressure Coolant Injection
IFI	-	Inspection Followup Item
ISEG	-	Independent Safety Engineering Group
NE	-	Nuclear Engineering
NEP	-	Nuclear Engineering Procedure
NOV	-	Notice of Violation
NPP	-	Nuclear Performance Plan
PCN	-	Project Change Notice
PI	-	Project Instruction
PMEN	-	Partial Modification Evaluation Notice
QA	-	Quality Assurance
QC	-	Quality Control
RCIC	-	Reactor Core Isolation Cooling
RHR	-	Residual Heat Removal
RPT	-	Recirculation Pump Trip
RPS	-	Reactive Protection System
RTS	-	Return to Service
SAR	-	Safety Evaluation Report
SLC	-	Standby Liquid Control
SDSP	-	Site Director Standard Practice
TACF	-	Temporary Alteration Control Form
TOL	-	Thermal Overload
TSD	-	Task Scooping Document
TVA	-	Tennessee Valley Authority
URI	-	Unresolved Item
USQ	-	Unreviewed Safety Question
USQD	-	Unreviewed Safety Question Determination
UFSAR	-	Updated Final Safety Analysis Report
W-DCN	-	Nuclear Engineering Instituted Design Change Notice
WP	-	Work Package