U. S. NUCLEAR REGULATORY COMMISSION
REGION II

Docket Nos:	50-327, 50-328
License Nos:	DPR-77, DPR-79
Report No:	50-327/96-16, 50-328/96-16
Licensee:	TVA
Facility:	Sequoyah Units 1 & 2
Location:	Sequoyah Access Road Hamilton County, TN 37379
Dates:	September 23-27, 1996; November 4-19, 1996 and December 16-19, 1996
Inspectors:	C. Smith, Reactor Inspector N. Merriweather, Reactor Inspector
Approved by:	C. Casto, Chief, Engineering Branch Division of Reactor Safety

Enclosure 2

EXECUTIVE SUMMARY
Sequoyah Nuclear Plant, Units 1 & 2
NRC Inspection Report 50-327,328/96-16

This special inspection included detail reviews of corrective actions implemented for Problem Evaluation Reports (PERs) No. SQP900372PER, Nuclear Fuel Design Changes Not Reconciled/Reflected in Design Basis Documents (DBDs); and SQ950021PER, Obtain Operability Evaluation for SQNP: Review of WBPER940576. Additionally, a review of the licensee's transition plans for implementing the EQ program after Phase 1 site engineering re-organization had been completed was performed. An Unresolved Item involving inadequate safety assessment of Rod Control System plant modification was closed and a Violation of 10 CFR 50 Appendix B, Criterion III was cited.

Results:

- An unresolved item concerning performance of an inadequate 10 CFR 50.59 Safety Evaluation that resulted in an Unreviewed Safety Question.
- An unresolved item concerning untimely revision to the EQ Binders and EEB calculations.
- An unresolved item concerning inadequate design control for "Nonconforming Plant Conditions."
- An unresolved item concerning the technical acceptability of reducing the calculated free field beta dose both inside containment and the annulus by 50 percent.
- An inspector followup item concerning inconsistent FSAR descriptions of the reactor power level.
- A violation for inadequate design controls for Rod Control System plant modification.

Report Details

III. Engineering

E1 Conduct of Engineering

E1.1 PER No. SQP900372PER, Nuclear Fuel Design Changes not Reconciled/Reflected in Design Basis Documents(DBDs)

a. Inspection Scope

The inspector reviewed PER No. SQP900372PER in order to evaluate the adequacy of the licensee's root cause analysis, extent of condition evaluation, and developed corrective actions for 10 CFR 50.49 identified deficiencies.

b. Observations and Findings

Condition Adverse to Quality Report (CAQR) SQP900372PER, dated September 18, 1990, documented fuel related design changes made by TVA which had not been reconciled or reflected in design basis documents. An increase in the average core burnup from 650 EFPD to 1000 EFPD resulted in an increase in the amount of core activity that is assumed at the start of a design basis LOCA. Because of this there was an increase in the 100 day integrated accident dose that electric equipment important to safety and qualified to 10 CFR 50.49 must withstand. TVA management prepared a Justification for Continued Operation (JCO), (TVA-91-293), to demonstrate that the requirements of 10 CFR 50.49 were still being met by equipment that had previously been environmentally qualified based on a source term of 650 EFPD. The inspector reviewed the JCO and determined that TVA had concluded that the JCO bounded reactor core designs with U235 fuel having average enrichment less than 4.5 percent 1000 EFPD burnup.

The NRC in a letter dated November 30, 1993, Subject: Evaluation of Increased Fuel Burnup on Equipment Qualification Sequoyah Nuclear Plant Unit 1 and 2, transmitted the results of the staff's review of the above JCO to TVA. The staff concluded that the JCO was not appropriate and TVA was requested to perform a reassessment of equipment qualification for 1000 EFPD burnup using an acceptable source term (TID-14844) and resubmit the JCO for the staff's review. TVA performed the reassessment and in a letter dated March 4, 1994, transmitted the JCO for the staff's review. The NRC in a letter dated April 8, 1994, informed TVA that the staff had reviewed the reassessment and determined that it satisfactorily responded to the staff's concern.

The inspector reviewed the results of the EQ reassessment titled "Review of 1000 EFPD with 4.5% U235 Enrichment", performed in support of the JCO submitted to the NRC. Corrective action plans developed and implemented for CAQR No. SQF870012, and SQP870165 were also reviewed during this inspection. The specific issues reviewed and the results of these reviews are discussed in the paragraphs below.

Technical Adequacy of 10 CFR 50.59 Safety Evaluation

CAQR No. SQF870012 was written on March 19, 1987, to document a condition where the core average exposure limit of 26154 MWD/MTU specified in FSAR Table 15.1.7-1 would be exceeded in Unit 1 cycle 4 operation. The suggested corrective action was to calculate the offsite dose using 1000 Effective Full Power Day (EFPD) and revise the FSAR to reflect the results of the revised calculation. CAQR No. SQP870165 was written to document the results of EGTS tests which demonstrated slow response of the dampers to pressure changes and missing design criteria which specified what the response time should be. The apparent cause of the dampers slow response to pressure changes was due to the use of a pressure indicating controller having only a proportional band with no reset function. The inspectors reviewed a 10 CFR 50.59 Safety Evaluation dated December 2, 1987, prepared by the licensee to make changes to the FSAR for resolution of the above deficiencies. Based on this review the inspectors determined that the following tables in the FSAR were being revised; 1) Table 15.1.7-1, Core and Gap Activities Based on Full Power Operation for 650 Days Full Power: 3565 MWt; 2) Table 15.5.3-3, Emergency Gas Treatment System Flow Rates; 3) Table 15.5.3-4, Offsite Doses From Loss of Coolant Accident; 4) Table 15.5.3-7, Control Room Personnel Doses for DBA Post Accident Period. Additionally, changes were being made to selected portions of the narrative descriptions in the FSAR to facilitate resolution of CAQR Nos. SQF870012 and SQP870165.

FSAR Chapter 15, Table 15.1.7-1 was revised to show new source terms based on 1000 EFPD operation. The results of offsite dose calculations performed by the NRC in support of licensing actions were documented in Safety Evaluation Report (SER) Supplement No. 1, dated February 1980. The inspectors reviewed section 15.4, of the SER to confirm if the FSAR changes and offsite dose analysis were acceptable and complied with the current licensing basis. One discrepancy was identified during this review. Offsite radiation doses contained in the SER Supplement No.1, Table 15-1, Radiological Consequences of Design Basis Accidents, was calculated by the NRC based on the assumption that Unit 1 reactor will be operated at a power level not in excess of 5% of the rated power of 3582 MWt. Table 15-2 of the SER, Assumptions Used in the Calculation of Loss of Coolant Accident Doses, also showed the reactor power level as 3582 MW thermal. This value of

reactor power level used in the offsite dose calculation was different from that used by TVA which was 3565 MW thermal. The guidance delineated in TID-14844, Calculation of Distance Factors For Power and Test Reactor sites, dated March 23,1962, requires the use of the reactor rated power level (megawatts) in the calculation which determines the radio nuclide inventory of specific isotopes. Numerous inconsistencies concerning the reactor rated power were identified in FSAR Tables 15.1.2-1; 15.1.7-1; and all the tables in FSAR section 15.5. The guaranteed core thermal power in Table 15.1.2-1 was listed as 3411MW thermal. In Table 15.1.7-1 it was listed as 3565 MW thermal and in all the tables of FSAR Section 15.5 it was listed as 3582 MW thermal. The maximum power level authorized in the facility operating license is 3411 MW thermal. This inconsistency in FSAR description of the reactor power level is identified as IFI 50-327,328/96-19-05, FSAR Inconsistent Description of Reactor Power Level.

The results of the above reviews demonstrated that the licensee had considered the consequences of offsite radiation doses to the health and safety of the public based on 1000 EFPD operation. Additional reviews of the 10 CFR 50.59 Safety Evaluation, however, revealed that the licensee had not evaluated whether the increase from 650 to 1000 EFPD operation affected the qualification status of equipment that had previously been qualified to a source term that was based on 650 EFPD criterion. The increase in EFPD from 650 to 1000 because of fuel related design changes had created an increase in the amount of core activity that was assumed at the start of a design basis LOCA. The increase in the core activity resulted in an increase in the 100 day integrated accident dose that environmentally qualified equipment must withstand. The licensing basis for the 10 CFR 50.59 EQ Program was 650 EFPD burnup and this requirement was exceeded by Unit 1 cycle 4 operation on December 29, 1989 and Unit 2 cycle 3 on December 30, 1988. This "Unreviewed Safety Question" involving failure of the 10 CFR 50.59 Safety Evaluation to address the requirements of environmentally qualified equipment resulted in nonconforming and unanalysed plant conditions from December 30, 1988 until July 30, 1990, when design basis Calculation TI-RPS-48, Integrated Accident Dose inside Containment and Annulus, Revision 3, was prepared to calculate the 100 day integrated accident dose based on the 1000 EFPD burnup criterion. This item is identified as unresolved item URI 50-327,328/96-16-01, Inadequate Safety Evaluation Resulted in Unreviewed Safety Question.

Corrective Actions Implemented for Nonconforming Plant Conditions

Problem Evaluation Report PER No. SQP900372PER was prepared on December 18, 1990, to document a condition where Nuclear Fuels (NF) made core design changes which had not been reconciled or reflected in

current Nuclear Engineering (NE) design basis documents. A "Cause Analysis" was performed for this deficiency and the apparent cause was determined to be lack of procedural controls to ensure adequate interface reviews and appropriate funding for those reviews. Corrective action plans developed and implemented for recurrence control included:

1. Revising Corporate Standard 9.2 for core alterations and core hardware changes to ensure adequate interface reviews and appropriate funding for these reviews.

2. Establishing requirements for NE to provide NF a list of fuel and core related parameters which affect engineering calculations and require review on a cycle specific basis.

3. Revising NF Instruction 3.0 to ensure that other design basis documents impacted by core component design changes were addressed.

Other corrective actions which required revising the EQ Binders and Electrical Engineering Branch (EEB) calculations to incorporate updated environmental conditions were delayed and transferred to PER No. SQ940040II, TROI action Item No. 36. The inspectors reviewed a copy of TROI Action Item No. 36 dated September 12, 1996, and verified that this item was still open.

The licensee prepared a JCO dated September 4, 1991, which was applicable to both Units and would permit continued operation until TVA revised the design documents to incorporate the 100 day integrated accident doses that were caused by the 1000 EFPD burnup criterion. TVA's JCO was based on the conclusions contained in a document titled "Tennessee Valley Authority, Sequoyah Nuclear Plants Units 1 and 2, Increase in the 100 Day Integrated Dose to Equipment in Containment Associated with Increased Fuel Burnup, Justification for Continued Operation." The JCO stated that TVA will reevaluate SQNP design basis following the NRC's final issuance of the new TID-14844 values in order to eliminate repetitive efforts of revising the EQ Binders.

On July 18, 1992, TVA management prepared JCO No. SQJCO92-013, Revision 0, and extended the time for implementing corrective actions related to TROI Action Item No. 36. This extension request was approved by the Site Vice-President on August 6, 1992. On September 17, 1993, JCO for SQP900372PER was extended by a corrective action request. The corrective action request was approved by TVA management on September 20, 1993.

Prior to preparation of the JCOs and during the intervals of time when TVA management postponed implementing the corrective action to revise the EQ Binders and EEB calculations, the core average exposure for both Units exceeded 650 EFPD operation on the dates listed:

Unit NO.	Cycle No.	Date EFPD Exceeded
1	4	12-29-89
1	5	06-09-91
1	6	11-29-92
1	7	04-02-95
2	3	12-30-88
2	4	05-24-90
2	5	09-28-91
2	6	01-03-94
2	7	10-05-95

On November 30, 1993, the NRC transmitted the results of their review of the "Westinghouse Technical JCO for SQNP" to TVA. TVA was informed that the JCO was technically inadequate and that it should be prepared in accordance with the guidelines of TID-14844. TVA was also requested to perform a reassessment of equipment qualification based on 1000 EFPD criterion using an acceptable source term and submit it to the NRC for their review. In response to this request on February 11, 1994, TVA prepared "JCO for PER No. SQP900372PER" which bounds reactor core designs with U235 average enrichment of less than 4.5% and 1000 EFPD. This JCO included Unit 2 cycle 6, Unit 1 cycle 7, and Unit 2 cycle 7 fuel cycle operation. The NRC reviewed "JCO for PER No. SQP900372PER"and concluded that TVA's equipment qualification reassessment satisfactorily responded to their concern. The results of this review was transmitted to TVA on March 4, 1994.

The inspectors determined that the licensee had continued plant operations under the JCO without revising the EQ Binders and EEB calculations. This untimely corrective action for revising the EQ Binders and EEB calculations is a concern and is identified as unresolved item URI 50-327,328/96-16-02, Untimely corrective action for nonconforming plant conditions.

Design Control Implemented for Nonconforming Plant Conditions

On July 30, 1990 TVA management approved design basis calculation TI-RPS-48, Integrated Accident Dose Inside Primary Containment and Annulus, Revision 3. This analysis was performed to determine the integrated accident doses inside the primary containment for equipment qualification based on the EFPD for calculating the equilibrium reactor core activity being increased from

650 EFPD to 1000 EFPD. The analysis was based on the assumption that core activity is instantaneously released (at t=0) within the primary containment in the following fractions of the core inventory.

100%	Noble Gases
50%	Iodines
50%	Cesium
1%	Other Fission Products

Revision 2 of this calculation used a burnup of 650 EFPD and was previously the calculation of record for demonstrating compliance with the requirements of 10 CFR 50.59. The results of Revision 3 of the calculation when compared to the 100 day integrated accident doses in revision 2 were as follows:

Location	TI-RPS-48, R2	TI-RPS-48-R3
Upper Containment		
Gamma	3.8 E7	3.0 E7
Beta	4.7 E8	8.3 E8
Instrument Rooms		
Gamma	1.048 E7	1.6 E7
Beta	4.7 E8	8.3 E8
Lower Containment		
Gamma	2.8 E7	2.5 E7
Beta	4.7 E8	8.3 E8
Accumulator and Fan Rooms		
Gamma	1.048 E7	1.6 E7
Beta	4.7 E8	8.3 E8
Raceway		
Gamma	1.048 E7	2.4 E7
Beta	4.7 E8	8.3 E8
Ice Condenser Bed		
Gamma	1.34 E7	2.3 E7
Beta	4.7 E8	8.3 E8
Annulus		
Gamma	1.3 E7	5.9 E6
Beta	5.0 E5	1.38 E6

A significant increase in free field Beta radiation resulted from the 1000 EFPD burnup criteria. The results of these calculations were never incorporated in Calculation TI-ECS-55, Summary of Harsh Environment Conditions for Sequoyah Nuclear Plant. As a consequence the environmental data drawings series Number 47E235 were never revised to reflect the integrated accident doses caused by the new source terms based on 1000 EFPD operation.

Additionally, FSAR Figures 3.11.2-1 and 3.11.2-2 were never revised to reflect the new 100 day integrated dose based on 1000 EFPD operation. The accident doses on the FSAR Figures were not consistent with the design basis of 1000 EFPD delineated in FSAR Table 15.1.7-1. This failure to control plant configuration and ensure that actual plant configuration is accurately depicted on drawings and has been reconciled with design basis is of concern and is identified as one example of unresolved item URI 50-327,328/96-16-03 Inadequate design control for "Nonconforming Plant" conditions.

On December 12, 1991, TVA management approved design basis calculation TI-RPS-48, Revision 5, "Integrated Accident Dose Inside of Primary Containment and Annulus," to document the 100 day integrated accident dose based on 650 EFPD burnup criteria. The calculation was prepared to implement TVA's management decision to temporarily reduce the 1000 EFPD burnup criterion. Calculation TI-ECS-55, Revision 16 was prepared to incorporate and clarify usage of the Containment Buildings design basis post accident radiation doses determined from calculation TI-RPS-48, Revision 5. Additionally, plant modification DCN No. 508114A, Revision 16, revised environmental drawing sheets 45, 47, and 48 to replace radiation values that were no longer conservative. The inspectors concluded that these drawing revisions were not an accurate representation of actual plant configuration based on FSAR Amendment 5 to table 15.1.7-1 which delineated 1000 EFPD. On June 9, 1991, Unit 1 cycle 5 operation exceeded the 650 EFPD burnup criterion that was being used as the basis for the 100 day integrated accident doses shown on the environmental drawings. This event was preceded by Unit 1 cycle 4 and Unit 2 cycles 3, 4 and 5 having average core exposure in excess of 650 EFPD. TVA's management failure to control plant configuration and ensure that actual plant configuration is accurately depicted on drawings and has been reconciled with design basis is of concern and will be identified as another example of unresolved item URI 50-327,328/96-16-03.

On March 4, 1994, TVA transmitted "JCO for PER No. SQP900372," dated February 11, 1994, to the NRC for their review. One hundred day integrated gamma, and beta accident doses for the 1) the upper containment; 2) lower containment; 3) Accumulator Fan Instrument Rooms; 4) Raceway; 5) Ice Bed Condenser and 6) Annulus were listed in the JCO. The inspectors reviewed the JCO and determined that the radiation values delineated in the JCO were

not supported by an approved analysis. A formal calculation had never been prepared, reviewed and approved to determine the 100 day integrated accident dose inside the containment and the annulus. The inspectors expressed concern to TVA management concerning the apparent non-compliance with the requirements of the design control program which requires that design analyses shall be performed in a planned, controlled, and correct manner. In response to the inspector's concern TVA attempted to reconstitute the analysis via computer runs on November 7, 1996. The raw computer data that resulted from this effort was not comprehensible to the inspectors. Calculation No. SBNSQS2-0163, Dose in Containment and Annulus with 1000 EFPD Burnup and 4.5 percent U235 Enrichment, was finally prepared and approved on November 15, 1996 to address the inspector's concern. The results of this calculation were reviewed by the inspectors and were determined to be comparable to the 100 day integrated accident doses for 1000 EFPD at 4.5 percent U235 listed in the JCO. TVA's failure to comply with the requirements of the design control program concerning engineering analyses is of concern and will be identified as one example of URI 50-327,328/96-16-03. Inadequate design control for Nonconforming Plant Condition.

Technical Acceptability of Reducing Calculated Free Field Beta Dose by 50 Percent

Design Calculation SQN-TI-RPS-048, Revision 6 issued October 1994, is the design basis calculation for the maximum 100-day integrated doses inside containment and the annulus with source terms for power levels of 3565 MWt. with average core burnups of 1000 EFPD and enrichments of 5 percent weight U235. The maximum free field Beta dose in air inside containment was calculated to be 6.311E+8 rads over 100 days. The licensee then made the assumption that the maximum calculated free field Beta dose could be reduced by a factor of 1/2 to account for a semi-infinite source geometry due to component self-shielding effects. The 50 percent reduction resulted in a surface Beta dose of 3.156E+8 rads that was below the previously analyzed Beta Dose given in Revision 2 of the calculation at 650 EFPD and 3565 MWt. using TID 14844 source terms. NUREG 0588, For Comment Version and Revision 1, Section 1 contains positions related to the establishment of the service conditions for areas inside and outside containment to which equipment should be qualified. It includes guidance for determining the radiation environments expected to occur during a design basis event condition. In Section 1.4(7), Radiation Conditions Inside and Outside Containment, it requires that the maximum Beta dose at the surface of unshielded equipment be taken as the free field Beta dose calculated for a point at the containment center. The licensee did not follow this guidance when they took the 50 percent reduction for self-shielding. The licensee indicated that this 50 percent reduction is standard industry practice and has

been previously accepted by NRC. The inspector acknowledged the licensee's position on this concern and indicated that this issue was unresolved pending further review by NRC.

The acceptability of the licensee reducing the calculated free field Beta Dose both inside containment and the annulus by 50 percent is unresolved and will be identified as URI 50-327,328/96-16-04.

c. Conclusion

The inspectors concluded that the licensee failed to implement adequate design controls for reactor core design changes and failed to take prompt and effective corrective action for nonconforming plant conditions identified since September 18, 1990. Three violations were identified. Additional review by the NRC has resulted in these violations being changed to URIs pending additional NRC reviews. One unresolved item and one inspector followup item was also identified.

E2 Engineering Support of Facilities and Equipment

E2.1 PER No. SQ950021PER, Obtain Operability Evaluation for SQNP: Review of WBPER940576

a. Inspection Scope

The inspector reviewed PER No. SQ950021PER in order to evaluate the adequacy of the licensee's root cause analysis, extent of condition evaluation, and developed corrective actions for 10 CFR 50.49 identified deficiencies.

b. Observations and Findings

Watts Bar Adverse Condition Report WBPER940576 identified a problem with the pressurizer PORVs where the energized times did not agree with limitations imposed by the EQ program. The PORVs had been energized in excess of 200 hours per year via 56 cycles which exceeded energized times specified in the EQ binder. This issue was reviewed for applicability to Sequoyah. EQ binder SQNEQ-SOL-002 documents that the pressurizer PORVs are energized for a maximum of 40 hours per year. Investigation revealed, however, that the 40 year energization time documented in the EQ binder was nonconservative in that the Target Rock solenoid valves had been in use since 1983.

The root cause analysis performed by the licensee was reviewed by the inspector and was determined to have been adequately performed. Interim

corrective actions taken to address this issue involved completing an Operability Determination where it was concluded that the PORVs could perform satisfactorily until the cycle 7 outage. The pressurizer PORV solenoid valves were subsequently replaced during the cycle 7 refueling outage of each unit. Corrective action plans developed for final resolution of this issue involved a review of the SQNP EQ binders to determine if revisions were required for any EQ binder, and supporting Qualified Life, or Accident Degradation Equivalency Calculations. The results of this review identified 12 EQ binders that required revision. The inspector reviewed the status of corrective action C.9.8 and C.9.9 and determined that the Qualified Life and Accident Degradation Calculations had not been revised to reflect identified duty cycle/operational time changes. Additionally, revisions to EQ binders based on the results of the above calculations have been restrained because of failure to promptly complete the calculations.

c. Conclusions

The inspector concluded that the Operability Determination performed for PER No. SQ950021PER was technically adequate. Interim corrective actions of replacing the pressurizer PORV solenoid coils during cycle 7 refueling outage of each unit also demonstrated TVA's implementation of prompt corrective action. TVA's management failure, however, to complete corrective actions C.9.8 and C.9.9 for an issue identified on January 13, 1993 was considered less than timely.

E6 Engineering Organization and Administration

a. Inspection Scope

The inspector reviewed the licensee's program documents that control the environmental qualification program to verify 1) that responsibilities had been defined and 2) requirements had been specified for establishing and maintaining the auditable documentation demonstrating qualification of equipment in compliance with 10 CFR 50.49. The licensee's transition plans for implementing the EQ program after Phase 1 site engineering re-organization was also reviewed.

b. Observation and Findings

Procedure SSP-6.5, Electrical Equipment Environmental Qualification (EQ) Program, Revision 7, is the controlling procedure for implementing the EQ program at Sequoyah. Based on review of this procedure the inspector determined that the program controls clearly identified functional responsibilities and levels of authority for adequate implementation of the EQ

program. Training requirements for personnel engaged in EQ work activities were also clearly identified on Appendix I of this procedure. No deficiencies were identified with the procedural controls in SSP-6.5.

The inspector reviewed the licensee's transition plans for implementing the 10 CFR 50.49 program after Phase 1 reorganization of the site engineering and material section. The following documents were reviewed during this effort.

- Procedure SPP-9.2, Equipment Environmental Qualification (EQ) Program, Revision 0.

- Procedure NEP-5.12, Program Requirements For Equipment Qualification of Electrical Equipment in Harsh Environments, Revision 1.

- Mechanical Design Standard No. DS-M18.14.1, Design Standard for Environmental Qualification of Electrical Equipment in Harsh Environment, Revision 0.

The inspector also conducted interviews with personnel engaged in EQ work activities from the EE/NE discipline and Maintenance Planning and Technical (MP/T) section. The interviews were intended to assess the level of the worker's understanding of the EQ program requirements and to verify that EQ training requirements had been met. All personnel interviewed were knowledgeable of the EQ program requirements and had completed EQ training. No deficiencies were identified with the licensee's staff involved with EQ program activities.

At the time of the inspection procedure SPP-9.2 was in the process of being reviewed for approval by NE management for replacing SSP-6.5 upon completion of the Phase 1 site engineering reorganization. This is an upper tier program document that delineate EQ program controls to be implemented at Sequoyah, Browns Ferry and Watts Bar. Based on this review the inspector determined that SPP-9.2 failed to adequately establish program controls for successful implementation of the EQ program at Sequoyah. Ownership of the EQ program was not identified; functional responsibilities and levels of authority for implementing the program was not described; and the implementing instructions lacked clarity and specificity because of the upper tier nature of the procedure. The procedure also failed to identify training requirements for personnel involved with EQ work activities.

TVA management was informed of this inspection finding. In response TVA management told the inspector that they concurred with the findings and procedure SPP-9.2 would not be approved for replacing SSP-6.5 in its present

form. The inspector was also advised of personnel changes that would be implemented on October 1, 1996, for Phase 1 reorganization of the site engineering and materials section. On this date TVA management will have only one person who have completed EQ training in the I&C section which now has ownership of the EQ program. Similarly, one EQ trained person will be in the MP/T section to perform EQ duties. The licensee has essentially de-centralized the EQ program, disbanded the dedicated staff who performed EQ activities, and has now included in the position descriptions of engineering and MP/T personnel requirements for performing EQ duties.

c. Conclusion

The inspector concluded that the transition plan for implementing the EQ program after Phase 1 reorganization of the site engineering section was inadequate based on procedure SPP-9.2. Additionally, the number of trained personnel required for performing EQ duties after October 1, 1996, does not appear to be adequate based on the numerous large scale ongoing corrective actions presently being implemented for identified EQ deficiencies.

E.8 Miscellaneous Engineering Issues

E.8.1 Employee's Concern Program

a. Inspection Scope

The inspector reviewed implementation of the licensee's Employee Concern Program to verify that employee's concerns related to inadequacies in the 10 CFR 50.49 Environmental Qualification Program are promptly and adequately addressed by TVA management.

b. Observations and Findings

Numerous concerns have been expressed by TVA personnel during exit interviews concerning the adequacy of the 10 CFR 50.49 Environmental Qualification Program. The inspector reviewed the employee's concerns documented in the following Concerns Resolution Program, (CRP) Files and conducted discussion with the Concerns Resolution Staff Manager concerning implementation of the program.

- File No. ECP-96-SQ-903
- File No. ECP-96-SQ-918
- File No. ECP-96-SQ-922
- File No. ECP-96-SQ-927
- File No. ECP-96-SQ-928

- File No. ECP-96-SQ-991

Based on these discussion's the inspector determined that File No. ECP-96-SQ-992-F1 was prepared as a collector file for issues raised by employees during exit interviews concerning the adequacy of SQNP programs. The scope of the employee's concerns included inadequacies involving the 10 CFR 50.49 EQ Program; Leak Rate Testing; Appendix R; Q List, Vendor Manuals; and Technical Specification Testings. TVA management had already taken actions to address these concerns. The inspector reviewed Engineering Reorganization Assessment Report, NASQ 96023-phase 1, and verified that EQ concerns were addressed in this investigation. The report concluded that although there has been a significant reduction in SQN Engineering personnel, contingency plans and tasks reassignments have been developed to ensure responsibilities are adequately assumed by remaining site and/or contract personnel.

Additional EQ concerns raised by employees have been documented in File No. ECP-96-SQ-A07-F1. These issues among others have been identified as action items to be included for review in upcoming audits. The inspector was informed that the results of the Engineering Reorganization Assessment-Phase 2, scheduled for January, 1997, will also provide additional indepth investigation of EQ concerns raised by TVA employees.

c. <u>Conclusion</u>

The inspector concluded that employee's concerns are promptly addressed by TVA management. Concerns involving inadequacies in implementing the 10 CFR 50.49 EQ program have not yet been fully investigated to validate the employees specific concerns. It is the inspectors understanding that the investigations to be performed during phase two of the engineering reorganization assessment will satisfy this requirement.

E.8.2 (Closed) Unresolved Item (URI) 50-327,328/96-02-04, Omission of Surveillance Tests for Rod Control System.

URI 50-327,328/96-02-04, was identified in connection with plant modification DCN No. M11445A, Revision 0, that was developed and implemented for Unit 1 during cycle 7, refueling outage. The plant modification was intended to address safety concerns described in NRC Generic Letter (GL) 93-04, Rod Control System Failure and Withdrawal of Rod Control Cluster Assemblies, 10 CFR 50.54 (f).

The safety assessment performed for this plant modification was determined to be technically inadequate. Specifically, the Safety Assessment Checklist,

Appendix G, Item 22, incorrectly stated that there were no new credible failure modes associated with the hardware change. This error led to omission of requirements from the DCN for development and implementation of recommend surveillances described in WCAP-13864, Revision 1. TVA management in their letter dated June 10, 1990, committed to the corrective action delineated in PER No. SQ960677 PER for developing a new procedure to comply with GL 93-04 and the WOG recommendations. The action due date for this corrective action is February 15, 1997. Additionally, plant modification DCN No. M11730A, has been revised to address the new failure modes introduced by the hardware changes. Based on the corrective actions completed by the licensee this URI is closed.

An apparent violation of 10 CFR 50 Appendix B, Criterion III, will be identified for failure to implement adequate design controls for "Rod Control System" plant modification.

C. Exit

The inspection scope and results were summarized with those persons indicated in paragraph D on November 22, 1996 and December 19, 1996. The inspector described the areas inspected and discussed in detail the inspection results. One unresolved item related to the technical acceptability of reducing the free field beta dose inside the containment and annulus by 50 percent was identified; and one inspector followup item concerning inconsistent FSAR description of the Reactor power was also identified. An Unresolved Item in connection with inadequate safety assessment of Rod Control System plant modification was closed, and a violation of 10 CFR 50 Appendix B, Criterion III was opened.

On January 8, 1997, in a telephone conversation, the licensee was informed that three unresolved items related to PER No. SQP900372PER were made unresolved items pending the results of a meeting with TVA. A date for the meeting was not yet determined.

Licensee Employees

R. Adney, site Vice President
*B. Alsup, Quality Assessment Supervisor
J. Beasley, Site Quality Manager
*L. Bryant, Assistant Plant Manager
*G. Buchanan, Component Engineering Manager
C. Butcher, Electrical Design Manager
M. Burzynski, Engineering and Materials Manager

*R. Driscoll, Site Training Manager
M. Fecht, Nuclear Assurance and Licensing Manager
T. Flippo, Site Support Manger
*J. Herron, Plant Manager
*C. Kent, Radchem Manager
*B. Lagergren, Operations Manager
*P. Leahy, Shift Manager, Operations
M. Lorek, Mechanical Engineering Manager
R. Newby, Concerns Resolution Staff, Manger
R. Norton, SQN Assessment Supervisor
R. Profitt, Licensing Engineer
J. Rupert, Engineering and Service Support Manager
*R. Shell, Licensing and Industry Affairs Manager
J. Smith, Site Licensing Supervisor

*Attended exit interview on December 19, 1996 only.

Inspection Procedures Used

IP 37550 Engineering
IP 37551 Onsite Engineering

Items Opened/Closed/Discussed

Opened

50-327,328/96-16-01	URI	Inadequate safety evaluation resulted in Unreviewed Safety Question. (Paragraph E1)
50-327,328/96-16-02	URI	Untimely corrective action for nonconforming plant conditions. (Paragraph E1)
50-327,328/96-16-03	URI	Inadequate design control for nonconforming plant conditions. (Paragraph E1)
50-327,328/96-16-04	URI	Technical acceptability of reducing the calculated free field beta dose inside containment and annulus. (paragraph E1)

50-327,328/96-16-05	IFI	FSAR inconsistent description of reactor power level. (Paragraph E1)
50-327/96-16-06	VIO	Inadequate Design Controls for Rod Control System plant modification. (Paragraph E.8)

Closed

URI 50-327,328/96-02-04	Omission of Surveillance Tests for Rod Control System.

Acronyms

CAQR	Condition Adverse to Quality Report
CFR	Code of Federal Regulations
DBDs	Design Basis Documents
EEB	Electrical Engineering Branch
EFPD	Effective Full Power Day
EGTS	Emergency Gas Treatment System
EQ	Environmental Qualification
FSAR	Final Safety Analysis Report
JCO	Justification for Continued Operation
LOCA	Loss of Coolant Accident
MWt	Megawatts Thermal
NE	Nuclear Engineering
NF	Nuclear Fuels
NRC	Nuclear Regulatory Commission
PER	Problem Evaluation Report
PORV	Power Operated Relief Valve
SER	Safety Evaluation Report
TVA	Tennessee Valley Authority
URI	Unresolved Item
WOG	Westinghouse Owners Group