UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

FACILITY OPERATING LICENSE

License No. DPR-77

1. The Nuclear Regulatory Commission (the Commission) having found that:

   A. The application for licenses filed by the Tennessee Valley Authority complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Sequoyah Nuclear Plant, Unit 1 (the facility), has been substantially completed in conformity with Provisional Construction Permit No. CPPR-72 and the application, as amended, the provisions of the Act and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter I;

   E. The Tennessee Valley Authority is technically and financially qualified to engage in the activities authorized by this operating license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. The Tennessee Valley Authority has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements", of the Commission's regulations;

   G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

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(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 220 are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications. R224

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

c. Performance of any test at a power level different from there described; and

NUREG-0011

Safety Evaluation Report

related to operation of

Sequoyah Nuclear Plant Units 1 and 2

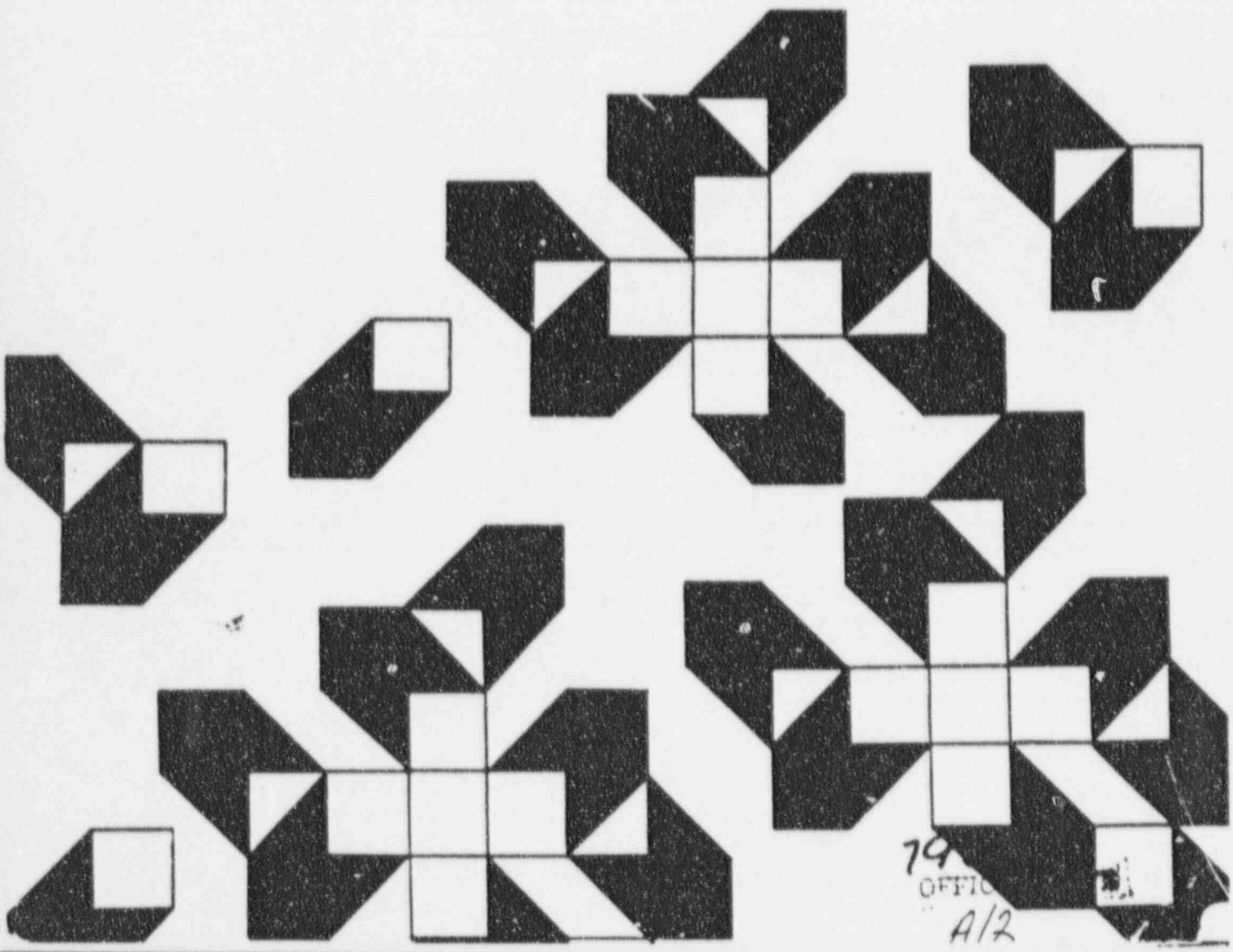
Tennessee Valley Authority

U. S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

Docket Nos. 50-327
50-328

March 1979



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stainless steels, constitutes an acceptable basis for meeting the requirements of General Design Criterion 26 of Appendix A to 10 CFR Part 50.

The design procedures and criteria that the applicant has used for the reactor internals are in conformance with established technical procedures, positions, standards, and criteria as cited above which are acceptable to the staff.

4.3 Nuclear Design

The nuclear design of Sequoyah Units 1 and 2 is the same as that of Trojan and Salem Unit 1. These reactors have been previously reviewed and approved. Sequoyah has rated power of 3411 thermal megawatts and consists of 193 assemblies containing the Westinghouse 17x17 rod fuel assembly array. Our review was based on information supplied by the applicant in the Final Safety Analysis Report and amendments thereto, and referenced topical reports. Our review was conducted within the guidelines provided by the Standard Review Plan, Section 4.3.

4.3.1 Design Bases

Design bases are presented which comply with the applicable General Design Criteria. Fuel design limits are specified which meet the requirements of General Design Criterion 10. A negative prompt feedback coefficient is required which satisfies General Design Criterion 11, and power oscillation is required either to be not possible or to be detected and suppressed by the control system, which satisfies General Design Criterion 12. A monitoring and control system is provided which automatically initiates a rapid reactivity insertion to prevent exceeding fuel design limits in normal operation and anticipated transients. This satisfies General Design Criteria 13 and 20. The control system is designed so that no single failure or single operator error will cause a violation of fuel design limits and so that shutdown is assured even when the single rod cluster control assembly (control rod) of highest worth is assumed to be stuck out of the core. Further a chemical shim system is provided which is capable of controlling normal power changes and bringing the reactor to cold shutdown. The control system, when combined with the engineered safety features, is required to control reactivity changes during accident conditions. Reactivity insertion rates and amounts are controlled so that limited damage occurs to the pressure boundary and the core stays in coolable geometry. The reactivity control system meets the requirements of General Design Criteria 25, 26, 27 and 28. On the basis of the above, we find the design bases presented in the Final Safety Analysis Report to be acceptable.

Design Description

The Final Safety Analysis Report contains the description of the first cycle fuel loading which consists of three different enrichments and has a first cycle of approximately one year. The enrichment distribution, burnable poison distribution,

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303

January 13, 1997

Tennessee Valley Authority
ATTN: Mr. Oliver D. Kingsley, Jr.
President, TVA Nuclear and
Chief Nuclear Officer
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: NOTICE OF VIOLATION
(NRC INSPECTION REPORT NO. 50-327/96-16 AND 50-328/96-16)

Dear Mr. Kingsley:

An NRC inspection was conducted on September 23-27, 1996, November 4-22, 1996, and December 16-19, 1996, at your Sequoyah facility. The purpose of the inspection was to determine whether activities authorized by the license were conducted safely and in accordance with NRC requirements. At the conclusion of the inspection the findings were discussed with those members of your staff identified in the enclosed report.

Areas examined during the inspection are identified in the report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel,and observation of activities in progress.

Based on the results of this inspection, certain of your activities appeared to be in violation of NRC requirements, as specified in the enclosed Notice of Violation (Notice). The violation is of concern because it is indicative of inadequate implementation of your design control program. Four unresolved items were also identified in connection with the use of high burnup fuel having average core exposure of 1000 Effective Full Power Days (EFPD). We are requesting a meeting with TVA to obtain additional information for resolution of these unresolved items.

The responses directed by this letter and the enclosed Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96-511.

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In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

Sincerely,

Original signed by
Charles A. Casto

Charles A. Casto, Chief
Engineering Branch
Division of Reactor Safety

Docket Nos. 50-327, 50-328
License Nos. DPR-77, DPR-79

Enclosures: 1. Notice of Violation
2. NRC Inspection Report

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COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

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NOTICE OF VIOLATION

Tennessee Valley Authority
Sequoyah Nuclear Plant

Docket Nos. 50-327 and 50-328
License Nos. DPR-77 and DPR-79

During NRC inspections conducted on September 23-27, 1996, November 4-22, 1996, and December 16-19, 1996, a violation of NRC requirement was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> 10 CFR 50, Appendix B, Criterion III requires that measures shall be established to ensure that design activities shall be prescribed and accomplished in accordance with procedures of a type sufficient to assure that applicable design inputs are correctly translated into specifications, drawings, procedures, or instructions. Applicable design inputs such as design bases, regulatory requirements, codes and standards shall be identified, documented, and their selection reviewed and approved. The design input shall be specified on a timely basis and to a level of detail necessary to permit the design activity to be carried out in a correct manner and to provide a consistent basis for making design decisions, accomplishing design verification measures, and evaluating design changes.
>
> Contrary to the above plant modification DCN No. M11730A, Revision 0, was approved on November 30, 1995, for modifying Unit 1 Rod Control System without incorporating applicable design inputs concerning new failure modes introduced by the hardware modification described in Westinghouse Topical Report WCAP-13864, Section 3.5. This failure resulted in the plant modification package omitting requirements for development and implementation of new surveillance tests required to determine any component failure which is undetectable during normal operation.

This is a Severity Level IV Violation (Supplement I)

Pursuant to the provisions of 10 CFR 2.201, Tennessee Valley Authority is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region II, and a copy to the NRC Resident Inspector, Sequoyah Nuclear Plant, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Because your response will be placed in the NRC Pubic Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. However, if you find it necessary to include such information, you should clearly indicate the specific information that you desire not to be placed in the PDR, and provide the legal basis to support your request for withholding the information from the public.

Dated at Atlanta, Georgia
this 13th day of January 1997

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

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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 50-327,328/96-10-05, 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 interf[illegible]e reviews and appropriate funding for those reviews. Corrective actio[illegible]ans 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.

<u>Technical Acceptability of Reducing Calculated Free Field Beta Dose by 50 Percent</u>

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. <u>Conclusions</u>

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. <u>Inspection Scope</u>

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. <u>Observation and Findings</u>

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. Conclusion

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

Safety Evaluation Report

related to operation of

Sequoyah Nuclear Plant Units 1 and 2

Tennessee Valley Authority

Supplement No. 1

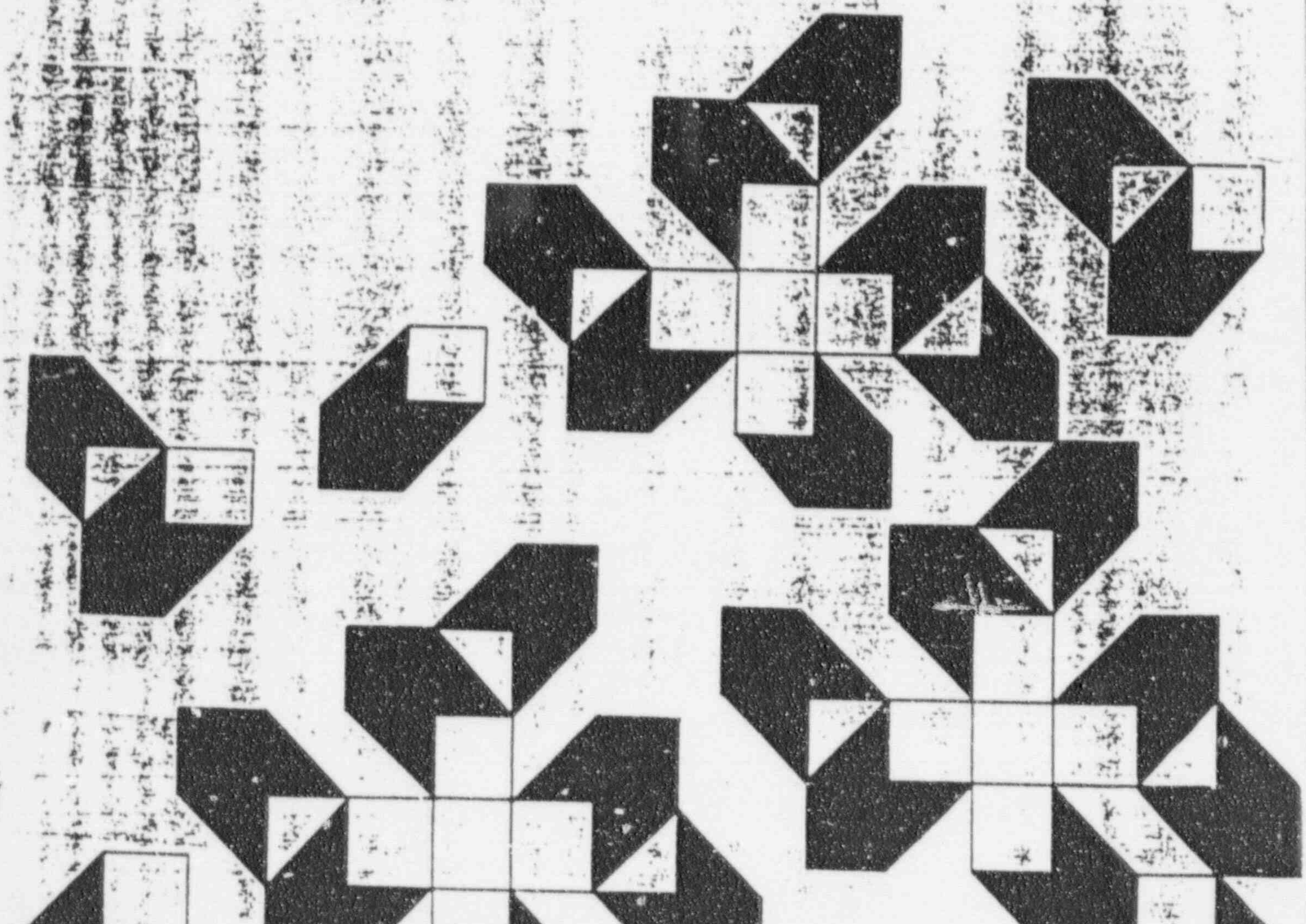
NUREG-0011
Supp. 1

U. S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

Docket Nos. 50-327
50-328

February, 1980



A/3

15.0 ACCIDENT ANALYSIS

15.2 Normal Operation and Anticipated Operational Transients

Boron Dilution

In the Safety Evaluation Report we stated the reliance upon an audible rate count to alert the operator of postulated boron dilution events during refueling was not justified.

The applicant provided justification for maintaining the alarm setpoint within one-half decade of the source flux level. Based on this margin and on the maximum possible rates of dilution, the applicant's analysis showed that the event would be detected and announced by the high flux at shutdown alarm within a time period that left sufficient margin for the operator to correct the situation before criticality occurred. Fifteen minutes is the required minimum time margin at these conditions in accordance with our Standard Review Plan.

The applicant has committed to a schedule for setting and monitoring the gap between the high flux at the shutdown alarm level and the shutdown source flux level that is consistent with the analysis presented. The setting is to be no higher than 1/2 decade above the count rate, and the margin is to be verified (or reset if necessary) every 30 minutes for the first 2 hours, every 2 hours for the next 6 hours, and once per shift thereafter until the flux level has stabilized. The required procedures and schedule for verification of the setpoint are to be incorporated in the operator's Surveillance Instructions.

The staff finds that the analysis, the reactivity changes in the boron dilution event are accounted for satisfactorily. The applicant's analysis defines a region of reactor conditions for the event that are considered safe, according to NRC criteria as described in SRP Section 15.4.6. The procedures adopted by the applicant will assure that the reactor remains within the boundaries of the safe conditions. The staff, therefore, regards the question of the boron dilution event immediately following shutdown as having been satisfactorily resolved.

ATWS

We have reviewed the TVA submittal of October 17, 1979, on Emergency Operating Procedures for the postulated anticipated transients without scram (ATWS) events. We provided our comments on the proposed procedures and made recommendations for changes. The proposed procedures must be modified in accordance with our comments and instructions to be acceptable for full power operation. However, the Sequoyah plant · · be operated at low power (less than or equal to five percent of full power prior to completion of procedures modifications without undue risk to the health and safety of the public. Our conclusion that low power operation is acceptable is based on our understanding of the expected plant response to the relevant ATWS events to occur under these operating conditions.

Normal Operation and Anticipated Operational Transients

Section 15.2 of the Sequoyah SER referred to our generic review of the Westinghouse Topical Reports WCAP-9226, WCAP-9236, and WCAP-9230 as the licensing bases for the analysis methods and sensitivity studies for postulated main steamline and feedline breaks. The steamline break information is contained in WCAP-9226. The feedwater line break information was provided in WCAP-9230 and in WCAP-9236, which discusses the NOTRUMP computer program used in the analyses. At that time, our review was scheduled for completion in late 1979.

For review of the steamline break topical, the staff requested additional information from Westinghouse in September 1978. Westinghouse responded with answers to some of our questions in May 1979. In response to staff inquiries, Westinghouse has attributed their failure to answer the balance of our questions to higher priority TMI-2 analyses requirements.

The staff has previously accepted steamline and feedline break analyses described in plant applications for PWRs designed by Westinghouse and other reactor vendors. It has been our position that a more detailed account of analytical methods for steamline and feedline break is required from the vendors for generic review · that the outcome of this review would be applied to licensed reactors. Our g· · review includes the performance of in-house audit calculations and calculations by technical assistance contractors.

Based on our preliminary review, there is sufficient evidence to conclude that substantial thermal margin exists under postulated steamline and feedline break accident conditions to preclude core damage leading to unacceptable consequences. Therefore, we conclude that the steamline and feedline break accident analyses for Sequoyah are acceptable while our more detailed review continues. However, our approval is predicated on the assumption that our generic review can proceed on a reasonable schedule. To assure that this assumption is valid, we will require a response to our outstanding questions on the topical reports discussed above and a new commitment for prompt response to any additional information requirements prior to approval of a full power operating license.

15.3 Accidents and Infrequent Transients

15.3.3 Steam Line Break

Long-Term Effects of Steam Line Break

Because the primary system pressure may have an effect on pressure vessel integrity following a steamline break or a small break loss-of-coolant accident, the staff requested additional information regarding the long-term scenarios, and effects of these events. Using techniques similar to those reviewed and approved for the D. C. Cook, Unit 2, plant, the applicant has conservatively calculated pressure and temperature conditions for a bounding spectrum of steamline break and small break LOCA events.

Using fracture mechanics techniques the applicant has estimated that, for those accident conditions, reactor vessel integrity can be assured for 17 effective full-power years. The fracture mechanics analysis performed by the applicant is similar to those that we have reviewed for other plants. Although we have not formally accepted these analyses, we do believe they are reasonable and provide assurance that the Sequoyah Units 1 and 2 reactor vessels have adequate margin against failure under postulated accident conditions for a substantial number of years of operation.

As described in Appendix C, Generic Task A-11 is expected to result in an engineering method and safety criteria that will provide the basis for assessing the acceptability of operation over the life of the plant, for both normal transient and accident conditions including consideration of MSLB and small break LOCA. The results of Task A-11 are expected to be available long before they are needed to provide this assessment for the Sequoyah Units 1 and 2 reactor vessels.

Based on the foregoing we have concluded that there is reasonable assurance that the integrity of the Sequoyah Units 1 and 2 reactor vessels will be maintained during postulated accidents.

Auxiliary Feedwater Runout Flow Following a Steam Line Break

The applicant was requested to address the potential for containment overpressurization due to the anticipated continuous addition, at pump runout flow, of auxiliary feedwater to the affected steam generator following a postulated main steam line break (MSLB) accident.

Our interest in this issue resulted from the 10 CFR Part 21 deficiency report filed by the Virginia Electric and Power Company (VEPCO) dated September 4, 1979. In that report, the NRC was informed by VEPCO that overpressurization of the continment at North Anna, Units 3 and 4, could occur in the event of a postulated MSLB inside containment. VEPCO indicated that, due to the anticipated continuous addition of auxiliary feedwater to the broken loop steam generator, at the pump runout flow condition, following a MSLB accident, the containment pressure will reach the containment design pressure in about 10 minutes.

To determine if the issue under consideration was generic for all pressurized water reactors (PWRs), we initiated a review of all "near-term" operating license applications for PWR plants. The object of the review was to determine if auxiliary feedwater flow was considered in the MSLB analyses and, if so, whether pump runout flow conditions were used.

The applicant indicated that the auxiliary feedwater system utilizes runout flow control equipment to limit the flow. Therefore, in the original MSLB analysis, the auxiliary feedwater flow to the faulted steam generator was assumed to exist at maximum capacity from the time of the rupture until realignment of the system is completed by the operator, 10 minutes after the onset of the postulated accident. The applicant's original submittal, stated that in one of the postulated analyses performed, a failure of the auxiliary feedwater runout protection system was assumed. In this analysis, it was assumed that flow to the broken loop steam generator at pump runout flow conditions continued from onset of the accident until the operator manually terminates flow 10 minutes later. It was concluded by the applicant, and the staff concurs, that the peak containment pressure will remain below the containment design pressure. The applicant also indicated that information for use in deciding to terminate the auxiliary feedwater flow to the affected steam generato will be available to the operator immediately after onset of the accident.

Based on our review of the applicant's evaluation, we find that the applicant's analyses have correctly accounted for the auxiliary feedwater flow and that no further analysis is required.

Normal Operation and Anticipated Operational Transients

We have reviewed the TVA submittal of November 9, 1979 responding to IE Information Notice 79-22 on qualification of control systems for Sequoyah Units 1 and 2. The submittal identifies plant systems required for safety and states for each safety function that adequate instrumentation would alert the operator to an event, adequate time is available for operator action, and control system design permits operator action. Based on the information provided by the applicant, our review of the Sequoyah Final Safety Analysis Report, our related reviews of equipment qualification, and similar reviews for operating reactors, we have found no event sequence that leads to an unacceptable consequence.

We have concluded that the Sequoyah applicant has satisfied the standards set for operating reactors and that this issue presents no concerns which would restrict operation of the plant.

15.4 Radiological Consequences of Accidents

15.4.1 Loss-of-Coolant Accident

This section of the supplement revises in its entirety the material that was present in the Safety Evaluation Report. The Sequoyah Nuclear Plant includes a double containment design to collect and filter the leakage of fission products from a postulated design basis loss-of-coolant accident. The double containment consists of a free-standing steel primary containment vessel surrounded by a reinforced concrete shield building. The reinforced concrete auxiliary building is also a part of the secondary containment barrier. Leakage which enters the secondary containment is treated by either the emergency gas treatment system or the auxiliary building gas treatment system prior to release to the atmosphere. Both of these systems are engineered safety features. Another engineered safety feature is the ice condenser with a sodium tetraborate additive to the ice to enhance the removal of iodine in the containment following a loss-of-coolant accident. The dose model and dose conversion parameters are consistent with those given in Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

In the analysis of the design basis loss-of-coolant accident, the primary containment was assumed to leak at the design leak rate of 0.25 percent per day for the first 24 hours following the accident and at 0.125 percent per day thereafter. The applicant established to the staff's satisfaction that the shield building annulus pressure would not exceed -0.25 inch water gauge pressure and that no leakage would bypass the gas treatment system throughout the course of the accident (see Section 6.2 of this report for further discussion of these items). The applicant has increased the amount of leakage which enters the auxiliary building following the accident from 10 percent to 25 percent of the primary containment leakage, assuming that this leakage was exhausted directly to the atmosphere during the first 10 minutes of the accident. After 10 minutes the leakage is processed through the auxiliary building gas treatment system without credit for holdup or mixing.

Seventy-five percent of the leakage from the primary containment enters the shield building annulus where we assumed that it went directly to the intake of the shield building annulus recirculation/exhaust system. Following passage through the emergency gas treatment system filters, a fraction of this leakage was assumed in our analysis to be exhausted to the atmosphere with the remainder recirculated to the shield building annulus where credit was given for mixing in 50 percent of the annulus free volume. The split between the exhaust and recirculation fractions was assumed to be proportional to the air flow rates in the exhaust and recirculation paths of the systems.

The applicant assumed in his dose analysis that it takes 10 minutes to isolate the auxiliary building rather than the previous assumption of 5 minutes (the applicant's analysis of the auxiliary building gas treatment system indicated that the system is designed to draw down the building to a -0.25 inch water gauge pressure within 170 seconds). Therefore, our analysis assumes that all leakage into the auxiliary building for the first 10 minutes into the accident is immediately released to the environment. For all times after the first 10 minutes into the accident we assume the leakage is exhausted through the gas treatment system.

The doses we calculate for the postulated design basis loss-of-coolant accident for the Sequoyah Nuclear Plant, shown in Table 15-1, are within the exposure guidelines of 10 CFR Part 100.

As part of the loss-of-coolant accident, we have also evaluated the consequences of leakage of containment sump water which is circulated by the emergency core cooling system after that postulated accident. We have assumed the sump water contains a mixture of iodine fission products in agreement with Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident." During the recirculation mode of operation the sump water is circulated outside of the containment to the auxiliary building. If a source of leakage should develop, such as from a pump seal failure, a fraction of the iodine in the water could become airborne in the auxiliary building and exit to the atmosphere. Since the emergency core cooling system area in the auxiliary building is served by an engineered safety features air filtration system (the auxiliary building gas treatment system), we conclude that the doses resulting from the postulated leakage of recirculation water would be low and, when added to the direct leakage loss-of-coolant accident doses, would result in total doses that are within the guideline values of 10 CFR Part 100.

As discussed in Section 6.2.3 of this report, the applicant recently informed us that during the ongoing Unit 2 construction activities, the minimum pressure that can be achieved in some of the ESF pump rooms will be approximately -0.04 inches water gauge as compared to the -0.25 inches required by the Technical Specifications. We determined that this pressure is not sufficiently low to assure the removal of airborne iodine activity by the auxiliary building gas treatment system following a postulated accident. We, therefore, have evaluated the 30-day dose at the LPZ distance for a postulated ESF pump seal failure following a loss-of-coolant accident. We conservatively assumed no holdup, mixing or removal of the associated airborne iodine activity in the auxiliary building. We also assumed that the Unit 1 reactor will be operated during this interim period of unit construction at a power level not in excess of 5 percent of the rated power of 3582 Mw thermal. Other assumptions of our analyses are listed in Table 15.3.

Based on our evaluation we conclude that the radiological consequences associated with an ESF pump seal failure in conjunction with the doses resulting from a design basis accident are within the guidelines of 10 CFR Part 100. We also conclude that the Unit 1 reactor shall not be operated at a power level in excess of 5 percent of the rated power level unless the applicant can demonstrate, by [illegible] that the ESF pump room can achieve and maintain a pressure not higher than the -0.25 inch water gauge identified in the Technical Specifications.

The applicant may purge the containment periodically during reactor operation. Should a loss-of-coolant accident occur when the purge lines are open, a portion of the containment atmosphere plus a portion of any flashed reactor coolant containing radioactive iodine fission products would be released to the environment in the short interval before the purge isolation valves close and isolate the containment. We have estimated the radiological consequences of this event using conservative assumptions regarding the radioactive iodine concentration in the primary coolant, the amount of reactor coolant inventory released, and the flow rate through the valves. We conclude that the consequences are such that, even when added to the calculated doses from containment leakage, the total is within the guideline values of 10 CFR Part 100.

The applicant has provided redundant hydrogen recombiners for the purpose of controlling any accumulation of hydrogen within the primary containment following a loss-of-coolant accident. In the event of failure of both recombiners, the applicant has provided a backup system. The purged containment effluent would flow to the shield building annulus where it would be subsequently discharged to the atmosphere through the emergency gas treatment system filters. We find the combination of redundant recombiners plus a backup purge capability to be an acceptable method for controlling the potential contribution to the offsite doses from hydrogen purging following a loss-of-coolant accident.

While Unit 2 is under construction the equipment hatch of the Unit 2 containment building will be closed off from the interim auxiliary building by two steel roll-up doors. These doors must be closed in the case of an accident in order to draw down the interim auxiliary building to a negative pressure of 0.25 inch water gauge. These doors will be locked shut or alarmed in the Unit 1 control room under normal conditions and plant personnel will be stationed at the doors when they are in use in order to initiate their immediate closing in the case of an accident. The staff concludes that this control will provide adequate assurance that the interim auxiliary building can be drawn down to the required negative pressure.

TABLE 15-1

RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS

Accident	Exclusion Area* 2-Hour Dose, Rem Thyroid	Exclusion Area* 2-Hour Dose, Rem Whole Body	Low Population Zone** 30-Day Dose, Rem Thyroid	Low Population Zone** 30-Day Dose, Rem Whole Body
Loss of Coolant	194	9	28	1
Fuel Handling	20	1	<1	<1
Steam Line Break				
1) I-131 at 1 microcurie per gram	13	<0.1	<1	<0.1
2) I-131 at 60 microcuries per gram	26	<0.1	1	<0.1
Steam Generator Tube Rupture				
1) I-131 at 1 microcurie per gram	19	<0.1	1	<0.1
2) I-131 at 60 microcuries per gram	214	<0.1	10	<0.1
Control Rod Ejection				
1) Leakage through secondary side	42	<0.1	2	<0.1
2) Leakage through containment	97	<0.1	4	<0.1

Part 100 guideline dose values are: 300 rem thyroid
25 rem whole body

*Exclusion area minimum boundary distance = 556 meters
**Low population zone distance = 4828 meters

TABLE 15-2

ASSUMPTIONS USED IN THE CALCULATION OF LOSS-OF-COOLANT ACCIDENT DOSES

Power Level	3582 Megawatts thermal
Operating Time	3 years
Fraction of Core Inventory Available for Leakage	
Iodines	25 percent
Noble Gases	100 percent
Initial Iodine Composition in Containment	
Elemental	91 percent
Organic	4 percent
Particulate	5 percent
Primary Containment Volumes	
Upper Containment	7.16×10^5 cubic feet
Lower compartment (including ice condenser)	5.25×10^5 cubic feet
Shield Building Annulus Volume	3.75×10^5 cubic feet
Mixing Fraction in Annulus	50 percent
Annulus Ventilation Flow Distribution	

TABLE 15-2 (Con't)

Time Step	Recirculation Flow Cubic Feet Per Minute	Exhaust Flow, Cubic Feet Per Minute
0-46 seconds	0	0
46-200 seconds	500	3500
200-400 seconds	1500	2500
400-1000 seconds	3000	1000
1000 seconds - 30 days	3900	100
Filter Efficiencies		
Elemental Iodine		95 percent
Organic Iodine		95 percent
Particulate Iodine		95 percent
Ice Condenser Removal Efficiency Elemental Iodine		30 percent
Flow Rate through Ice Condenser		40,000 cubic feet per minute
Period of Ice Condenser Effectiveness		10-60 minutes
Primary Containment Leak Rates		
0 - 24 Hours		0.25 percent per day
> 24 Hours		0.125 percent per day
Bypassing Leakage Fraction (Auxiliary Building Pathway)		
0-10 Minutes		25 percent
>10 Minutes		0 percent
Minimum Exclusion Area Boundary Distance		556 meters
Low Population Zone Distance		4828 meters
Atmospheric Diffusion (X/Q) Values		
0-2 Hours		1.4×10^{-3} sec per cubic meter
0-8 Hours		6.4×10^{-5} sec per cubic meter
8-24 Hours		4.5×10^{-5} sec per cubic meter
1-4 Days		2.1×10^{-5} sec per cubic meter
4-30 Days		6.9×10^{-6} sec per cubic meter

TABLE 15-3

ASSUMPTIONS USED IN THE CALCULATION OF ESF PUMP SEAL FAILURE

Power Level	180 Megawatt thermal (5 percent of rated)
Atmospheric Diffusion Values	See Table 15-2
Liquid Volume in Primary Containment	500,000 gallons
Time of Pump Seal Failure After LOCA	24 hrs.
Pump Seal Failure Flowrate	60 gallons/minute
Isolation of Pump Seal Failure	30 minutes
Evaporation Fraction	0.1