NUREG-0847
Supplement No. 7

Safety Evaluation Report

related to the operation of
Watts Bar Nuclear Plant,
Units 1 and 2

Docket Nos. 50-390 and 50-391

Tennessee Valley Authority

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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ABSTRACT

This report supplements the Safety Evaluation Report (SER), NUREG-0847 (June 1982), Supplement No. 1 (September 1982), Supplement No. 2 (January 1984), Supplement No. 3 (January 1985), Supplement No. 4 (March 1985), Supplement No. 5 (November 1990), and Supplement No. 6 (April 1991) issued by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by the Tennessee Valley Authority, as applicant and owner, for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2 (Docket Nos. 50-390 and 50-391). The facility is located in Rhea County, Tennessee, near the Watts Bar Dam on the Tennessee River. This supplement provides recent information regarding resolution of some of the outstanding and confirmatory items, and proposed license conditions identified in the SER.

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ABBREVIATIONS

ACR	auxiliary control room
AIW	American Insulated Wire
ANL	Argonne National Laboratory
ASME	American Society of Mechanical Engineers
BFN	Browns Ferry Nuclear Plant
BISI	bypassed and inoperable status indication
BTP	branch technical position
CAP	corrective action program
CAQ	condition adverse to quality
CAQR	condition adverse to quality report
CCRS	computerized cable routing system
CFR	Code of Federal Regulations
CNPP	Corporate Nuclear Performance Plan (NUREG-1232, Vol. 1)
DCA	design control authorization
DECLG	double-ended cold leg
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
ECCS	emergency core cooling system
EQ	equipment qualification
ERCW	emergency raw cooling water
ESFAS	engineered safety features actuation system
FSAR	final safety analysis report
GDC	general design criterion
ICEA	Insulated Cable Engineers Association
IE	Office of Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
INPO	Institute of Nuclear Power Operations
LOCA	loss-of-coolant accident
MCR	main control room
MSLB	main steamline break
MSSV	main steam safety valve
NEC	National Electric Code
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OBE	operation basis earthquake
PORV	power-operated relief valve

QA	quality assurance
RCCA	rod cluster control assembly
RCS	reactor coolant system
RG	regulatory guide
RPS	reactor protection system
RTD	resistance temperature detector
RWST	refueling water storage tank
SER	safety evaluation report
SP	special program
SPDS	safety parameter display system
SQN	Sequoyah Nuclear Plant
SRP	Standard Review Plan
SRS	system requirement specification
SSER	supplement to SER
SSI	soil-structure interaction
SWBP	side wall bearing pressure
TAC	technical assignment control
TS	technical specifications
TSC	Technical Support Center
TVA	Tennessee Valley Authority
U-CONN	University of Connecticut
UHI	upper head injection
USI	unresolved safety issue
WBN	Watts Bar Nuclear Plant
WBNPP	Watts Bar Nuclear Performance Plan (NUREG-1232, Vol. 4)
WISP	Workload Information and Scheduling Program

1 INTRODUCTION AND DISCUSSION

1.1 Introduction

In June 1982, the Nuclear Regulatory Commission staff (NRC staff or staff) issued a Safety Evaluation Report, NUREG-0847, regarding the application by the Tennessee Valley Authority (TVA or the applicant) for licenses to operate the Watts Bar Nuclear Plant, Units 1 and 2. The Safety Evaluation Report (SER) was followed by Supplement No. 1 (SSER 1, September 1982), Supplement No. 2 (SSER 2, January 1984), Supplement No. 3 (SSER 3, January 1985), Supplement No. 4 (SSER 4, March 1985), Supplement No. 5 (SSER 5, November 1990), and Supplement No. 6 (SSER 6, April 1991).

The SER and SSERs were written in accordance with the format and scope outlined in the Standard Review Plan (SRP, NUREG-0800). Issues that arose as a result of the SRP review that were not closed out at the time the SER was published were classified into outstanding issues, confirmatory issues, and proposed license conditions (see Sections 1.7, 1.8, and 1.9, which follow).

In addition to the guidance of the SRP, the staff would from time to time issue generic requirements or recommendations in the form of bulletins and generic letters. Each of these bulletins and generic letters carries its own applicability, work scope, and acceptance criteria; some are applicable to Watts Bar. The implementation status was addressed in Section 1.14 of SSER 6. The staff is reevaluating the status of implementation of all bulletins and generic letters. Results of this reevaluation will be published in a future SSER.

Since SSER 4 was issued, Watts Bar licensing activities have been put on hold because of problems identified at TVA plants (see Section 1.13 for details). Thus, no supplements were issued in the ensuing five years. SSER 5 was issued in November 1990, signifying the staff's resumption of licensing activities. This supplement (SSER 7) provides more recent information regarding the resolution or status of some of the outstanding and confirmatory issues, and proposed license conditions identified in the SER and its supplements. Some of the issues addressed in previous SSERs may be subject to further review as a result of the corrective actions that the applicant is engaged in (see Section 1.13).

Each of the following sections or appendices of this supplement (SSER 7) is numbered the same as the section or appendix of the SER that is being updated, and the discussions are supplementary to, and not in lieu of, the discussion in the SER unless otherwise noted. Accordingly, Appendix A is a continuation of the chronology of the safety review. Appendix B is an updated bibliography.* Appendix E is a list of principal contributors to this supplement. Appendix G continues to note errata. In Appendix P, the staff's safety evaluation of April 25, 1991, is reproduced. This supplement made no changes in other appendices.

*Availability of all material cited is described on the inside front cover of this report.

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1.7 Summary of Outstanding Issues

SER Section 1.7 identified 17 outstanding issues (open items) that had not been resolved at the time the SER was issued. This SSER updates the status of some of those items and 5 new issues that were added. The current status of each of the issues is tabulated below and the relevant SER or SSER section is indicated. Those issues that are, to date, unresolved will be addressed in future SSERs.

Issue*	Status	Section
(1) Potential for liquefaction beneath ERCW pipelines and Class 1E electrical conduit	Resolved (SSER 3)	2.5.4.4
(2) Buckling loads on Class 2 and 3 supports	Resolved (SSER 4)	3.9.3.4
(3) Inservice pump and valve test program (TAC 74801)	Updated (SSER 5)	3.9.6
(4) Qualification of equipment		
(a) Seismic (TAC 71919)	Updated (SSER 6)	3.10
(b) Environmental (TAC 63591)	Under review (SER)	3.11
(5) Preservice inspection program (TAC 63627)	Under review (SER)	5.2.4, 6.6
(6) Pressure-temperature limits for Unit 2	On hold	5.3.2, 5.3.3
(7) Model D-3 steam generator preheater tube degradation	Resolved (SSER 4)	5.4.2.2
(8) Branch Technical Position CSB 6-4	Resolved (SSER 3); see License Condition 8	6.2.4
(9) H_2 analysis review	Resolved (SSER 4)	6.2.5
(10) Safety valve sizing analysis (WCAP-7769)	Resolved (SSER 2)	5.2.2

*The TAC (technical assignment control) number that appears in parentheses after the title is an internal NRC control number by which the issue is managed through the Workload Information and Scheduling Program and relevant documents are filed. Documents associated with each TAC number can be listed by the NRC document control system, NUDOCS/AD.

Issue	Status	Section
(11) Compliance of proposed design change to the offsite power system to GDC 17 and 18 (TAC 63649)	Under review (SSER 2, SSER 3)	8.2
(12) Fire protection program (TAC 63648)	Awaiting submittal (SER)	9.5.1
(13) Quality classification of diesel generator auxiliary system piping and components (TAC 63638)	Resolved (SSER 5)	9.5.4.1
(14) Diesel generator auxiliary system design deficiencies (TAC 63638)	Resolved (SSER 5)	9.5.4, 9.5.5, 9.5.7
(15) Physical Security Plan (TAC 63657)	Under review (SER)	13.6
(16) Boron-dilution event	Resolved (SSER 4)	15.2.4.4
(17) QA Program (TAC 76972)	Updated (SSER 5)	17
(18) Seismic classification of cable trays and conduit (TAC R00508, R00516)	Under review (SSER 6)	3.2.1, 3.10
(19) Seismic design concerns (TAC 79717, 80346):		
(a) Number of OBE events	Under review (SSER 6)	3.7.3
(b) 1.2 Multi-mode factor	Under review (SSER 6)	3.7.3
(c) Code usage	Under review (SSER 6)	3.7.3
(d) Conduit damping values	Under review (SSER 6)	3.7.3
(e) Worst case, critical case, bounding calculations	Under review (SSER 6)	3.7.3
(f) Mass eccentricities	Under review (SSER 6)	3.7.2.1.2
(g) Comparison of set A versus set B response	Opened (SSER 6)	3.7.2.12
(h) Category 1(L) piping qualification	Under review (SSER 6)	3.9.3
(i) Pressure relief devices	Resolved (SSER 7)	3.9.3.3
(j) Structural issues	Under review (SSER 6)	3.8
(k) Update FSAR per 12/18/90 letter	Opened (SSER 6)	3.7, 3.8, 3.9
(20) Mechanical systems and components (TAC 79718, 80345)		
(a) Feedwater check valve slam	Under review (SSER 6)	3.9.1
(b) New support stiffness and deflection limits	Under review (SSER 6)	3.9.3.4

Issue	Status	Section
(21) Removal of RTD bypass system (TAC 63599)	Under review (SSER 6)	5.1
(22) Removal of upper head injection system (TAC 77195)	Resolved (SSER 7)	6.3.1

In addition to the above 22 issues, the staff has, in the 6 years since SSER 4 was published, identified a number of new issues that require resolution as follows:

(23) Containment Isolation Using Closed Systems (TAC 63597)	Opened (SSER 7)	6.2.4
(24) Main Steam Line Break Outside Containment (TAC 63632)	Opened (SSER 7)	15.4.2
(25) Health Physics Program (TAC 63647)	Opened (SSER 7)	12.3, 12.5, 12.6, 12.7
(26) Regulatory Guide 1.97, Instruments To Follow Course of Accident (TAC 77550)	Opened (SSER 7)	7.5.2
(27) Containment Sump Screen Design Anomalies (TAC 77845)	Opened (SSER 7)	6.2
(28) Operating, Maintenance, and Emergency Procedures (TAC 77861)	Opened (SSER 7)	15.5.2

1.8 Confirmatory Issues

SER Section 1.8 identified 42 confirmatory issues for which additional information and documentation were required to confirm preliminary conclusions. This supplement updates the status of those items for which the confirmatory information has subsequently been provided by the applicant and for which review has been completed by the staff. The current status of each of the original issues is tabulated below, with the relevant SER or SSER section indicated. Resolution of issues that are outstanding, to date, will be addressed in future SSERs. Confirmatory Issue 43 was added in SSER 6.

(1) Design-basis groundwater level for the ERCW pipeline	Resolved (SSER 3)	2.4.8
(2) Material and geometric damping effect in SSI analysis	Resolved (SSER 3)	2.5.4.2
(3) Analysis of sheetpile walls	Resolved (SSER 3)	2.5.4.2
(4) Design differential settlement of piping and electrical components between rock-supported structures	Resolved (SSER 3)	2.5.4.3

Issue	Status	Section
(5) Upgrading ERCW system to seismic Category I (TAC 63617)	Resolved (SSER 5)	3.2.1, 3.2.2
(6) Seismic classification of structures, systems, and components important to safety (TAC 63618)	Resolved (SSER 5)	3.2.1
(7) Tornado-missile protection of diesel generator exhaust	Resolved (SSER 2)	3.5.2, 9.5.4.1, 9.5.8
(8) Steel containment building buckling research program	Resolved (SSER 3)	3.8.1
(9) Pipe support baseplate flexibility and its effects on anchor bolt loads (IE Bulletin 79-02) (TAC 63625)	Updated (SSER 6)	3.9.3.4
(10) Thermal performance analysis	Resolved (SSER 2)	4.2.2
(11) Cladding collapse	Resolved (SSER 2)	4.2.2
(12) Fuel rod bowing evaluation	Resolved (SSER 2)	4.2.3
(13) Loose-parts monitoring system	Resolved (SSER 3)	4.4.5
(14) Installation of residual heat removal flow alarm	Resolved (SSER 5)	5.4.3
(15) Natural circulation tests (TAC 63603, 79317, 79318)	Under review (SER)	5.4.3
(16) Atmospheric dump valve testing	Resolved (SSER 2)	5.4.3
(17) Protection against damage to containment from external pressure	Resolved (SSER 3)	6.2.1.1
(18) Designation of containment isolation valves for main and auxiliary feedwater lines and feedwater bypass lines (TAC 63623)	Resolved (SSER 5)	6.2.4
(19) Compliance with GDC 51	Resolved (SSER 4)	6.2.7, App. H
(20) Insulation survey (sump debris)	Resolved (SSER 2)	6.3.3
(21) Safety system setpoint methodology	Resolved (SSER 4)	7.1.3.1
(22) Steam generator water level reference leg	Resolved (SSER 2)	7.2.5.9
(23) Containment sump level measurement	Resolved (SSER 2)	7.3.2

Issue	Status	Section
(24) IE Bulletin 80-06	Resolved (SSER 3)	7.3.5
(25) Overpressure protection during low-temperature operation	Resolved (SSER 4)	7.6.5
(26) Availability of offsite circuits	Resolved (SSER 2)	8.2.2.1
(27) Non-safety loads powered from the Class 1E ac distribution system	Resolved (SSER 2)	8.3.1.1
(28) Low and/or degraded grid voltage condition (TAC 63649)	Updated (SSER 7)	8.3.1.2
(29) Diesel generator reliability qualification testing (TAC 63649)	Resolved (SSER 7)	8.3.1.6
(30) Diesel generator battery system	Resolved (SSER 2)	8.3.2.4
(31) Thermal overload protective bypass	Resolved (SSER 2)	8.3.3.1.2
(32) Sharing of dc and ac distribution systems and power supplied between Units 1 and 2 (TAC 63649)	Under review (SSER 3)	8.3.3.2.2
(33) Sharing of raceway systems between units	Resolved (SSER 2)	8.3.3.2
(34) Testing Class 1E power systems	Resolved (SSER 2)	8.3.3.5.2
(35) Evaluation of penetration's capability to withstand failure of overcurrent protection device (TAC 63649)	Resolved (SSER 7)	8.3.3.6
(36) Missile protection for diesel generator vent line (TAC 63639)	Resolved (SSER 5)	9.5.4.2
(37) Component cooling booster pump relocation	Resolved (SSER 5)	9.2.2
(38) Electrical penetrations documentation (TAC 63648)	Under review (SER)	9.5.1.3
(39) Compliance with NUREG/CR-0660 (TAC 63639)	Resolved (SSER 5)	9.5.4.1
(40) No-load, low-load, and testing operations for diesel generator (TAC 63639)	Resolved (SSER 5)	9.5.4.1
(41) Initial test program	Resolved (SSER 3)	14

Issue	Status	Section
(42) Submergence of electrical equipment as result of a LOCA (TAC 63649)	Under review (SER)	8.3.3.1.1
(43) Safety parameter display system	Updated (SSER 6)	18.2, App. P

1.9 Proposed License Conditions

In Section 1.9 of the SER and SSERs, the staff identified 43 proposed license conditions. Since these documents were issued, the applicant has submitted additional information on some of these items, thereby removing the necessity to impose a condition. The proposed license conditions are tabulated below, with the corresponding NUREG-0737 item number given in parentheses (as appropriate) and the relevant SER or SSER section indicated.

Proposed Condition	Status	Section
(1) Relief and safety valve testing (II.D.1)	Resolved (SSER 3)	3.9.3.3, 5.2.2
(2) Inservice testing of pumps and valves (TAC 74801)	Updated (SSER 5)	3.9.6
(3) Detectors for inadequate core cooling (II.F.2) (TAC 77132 and 77133)	Awaiting submittal (SER)	4.4.8
(4) Inservice Inspection Program (TAC 76881)	Under review (SSER 3)	5.2.4, 6.6
(5) Installation of reactor coolant vents (II.B.1)	Resolved (SSER 5)	5.4.5
(6) Accident monitoring instrumentation (II.F.1)		
(a) noble gas monitor (TAC 63645)	Resolved (SSER 5)	11.7.1
(b) iodine particulate sampling (TAC 63645)	Resolved (SSER 6)	11.7.1
(c) high-range in-containment radiation monitor (TAC 63645)	Resolved (SSER 5)	12.7.2
(d) containment pressure	Resolved (SSER 5)	6.2.1
(e) containment water level	Resolved (SSER 5)	6.2.1
(f) containment hydrogen	Resolved (SSER 5)	6.2.5
(7) Modification to chemical feedlines (TAC 63622)	Resolved (SSER 5)	6.2.4

Proposed Condition	Status	Section
(8) Containment isolation dependability (II.E.4.2) (TAC 63633)	Resolved (SSER 5)	6.2.4
(9) Hydrogen control measures (NUREG-0694, II.B.7) (TAC 77208)	Under review (SER)	6.2.5, App. C
(10) Status monitoring system/BISI (TAC 77136, 77137)	Resolved (SSER 7)	7.7.2
(11) Installation of acoustic monitoring system (II.D.3)	Resolved (SSER 5)	7.8.1
(12) Diesel generator reliability qualification testing at normal operating temperature	Resolved (SSER 2)	8.3.1.6
(13) dc monitoring and annunciation (TAC 63649)	Under review (SSER 3)	8.3.2.2
(14) Possible sharing of dc control power to ac switchgear	Resolved (SSER 3)	8.3.3.2.4
(15) Testing of associated circuits	Resolved (SSER 3)	8.3.3.3
(16) Testing of non-Class 1E cables	Resolved (SSER 3)	8.3.3.3
(17) Low-temperature overpressure protection/power supplies for pressurizer relief valves and level indicators (II.G.1) (TAC 63649)	Resolved (SSER 7)	8.3.3.4
(18) Testing of reactor coolant pump breakers	Resolved (SSER 2)	8.3.3.6
(19) Postaccident sampling system (II.B.3) (TAC 77543)	Updated (SSER 3, SSER 5)	9.3.2
(20) Fire protection program (TAC 63648)	Awaiting submittal (SER)	9.5.1
(21) Performance testing for communications systems (TAC 63637)	Resolved (SSER 5)	9.5.2
(22) Diesel generator reliability (NUREG/CR-0660) (TAC 63640)	Resolved (SSER 5)	9.5.4.1
(23) Secondary water chemistry monitoring and control program	Resolved (SSER 5)	10.3.4

Proposed Condition	Status	Section
(24) Primary coolant outside containment (III.D.1.1) (TAC 63646)	Updated (SSER 6)	11.7.2
(25) Independent safety engineering group (I.B.1.2) (TAC 63592)	Under review (SER)	13.4
(26) Use of experienced personnel during startup (TAC 63592)	Under review (SER)	13.1.3
(27) Emergency preparedness (III.A.1.1, III.A.1.2, III.A.2) (TAC 63656)	Under review (SER)	13.3
(28) Review of power ascension test procedures and emergency operating procedures by NSSS vendor (I.C.7) (TAC 77861)	Under review (SER)	13.5.2
(29) Modifications to emergency operating instructions (I.C.8) (TAC 77861)	Under review (SER)	13.5.2
(30) Report on outage of emergency core cooling system (II.K.3.17)	Resolved (SSER 3)	13.5.3
(31) Initial test program (TAC 79872)	Resolved (SSER 7)	14.2
(32) Effect of high-pressure injection for small-break LOCA with no auxiliary feedwater (II.K.2.13)	Resolved (SSER 4)	15.5.1
(33) Voiding in the reactor coolant system (II.K.2.17)	Resolved (SSER 4)	15.5.2
(34) PORV isolation system (II.K.3.1, II.K.3.2) (TAC 63631)	Resolved (SSER 5)	15.5.3
(35) Automatic trip of the reactor coolant pumps during a small-break LOCA (II.K.3.5)	Resolved (SSER 4)	15.5.4
(36) Revised small-break LOCA analysis (II.K.3.30, II.K.3.31) (TAC 77298)	Resolved (SSER 5)	15.5.5
(37) Detailed control room design review (I.D.1) (TAC 63655)	Updated (SSER 6)	18.1
(38) Physical Security Plan (TAC 63657)	Under review (SSER 1)	13.6.4
(39) Control of heavy loads (NUREG-0612) (TAC 77560)	Updated (SSER 3)	9.1.4

Proposed Condition	Status	Section
(40) Anticipated transients without scram (Generic Letter 83-28, Item 4.3) (TAC 64347)	Resolved (SSER 5)	15.3.6
(41) Steam generator tube rupture (TAC 77569)	Updated (SSER 3, SSER 5)	15.4.3
(42) Loose-parts monitoring system (TAC 77177)	Resolved (SSER 5)	4.4.5
(43) Safety parameter display system (TAC 73723 and 73724)	Opened (SSER 5)	18.2

1.11 Nuclear Waste Policy Act of 1982

Section 302(b) of the Nuclear Waste Policy Act of 1982 states that NRC shall not issue or renew a license for a nuclear power reactor unless the utility has signed a contract with the Department of Energy for waste disposal services.

By letter dated February 16, 1985, the applicant stated that it has such an agreement (Contract No. DE-CR01-83-NE 44420) with the Department of Energy. This agreement is applicable to both Watts Bar units.

1.12 Approved Technical Issues for Incorporation in the License as Exemptions

The applicant applied for exemptions from certain provisions of the regulations. These have been reviewed by the staff and approved in appropriate sections of the SER and SSERs. These technical issues are listed below and the actual exemptions will be incorporated in the operating license:

(1) Seal leakage test instead of full-pressure test (Section 6.2.6, SSER 4) (TAC 63615)

(2) Criticality monitor (Section 9.1, SSER 5) (TAC 63615)

1.13 Implementation of Corrective Action Programs and Special Programs

On September 17, 1985, the NRC sent a letter to the applicant, pursuant to Title 10 of the Code of Federal Regulations, Section 50.54(f), requesting that the applicant submit information on its plans for correcting problems with the overall management of its nuclear program as well as on its plans for correcting plant-specific problems. In response to this letter, TVA prepared a Corporate Nuclear Performance Plan (CNPP) that identified and proposed corrections to problems with the overall management of its nuclear program, and a site-specific plan for Watts Bar entitled, "Watts Bar Nuclear Performance Plan" (WBNPP). The staff reviewed both plans and documented results in two safety evaluation reports, NUREG-1232, Vol. 1 (dated July 1987), and NUREG-1232, Vol. 4 (dated January 1990).

NUREG-1232, Vol. 4, documented the staff's general review of most of the corrective action programs (CAPs) and special programs (SPs) through which the applicant would effect corrective actions at Watts Bar. When the report was published, some of the CAPs and SPs were in their initial stages of implementation. The staff stated that it will report its review of the implementation of all CAPs and SPs and closeout of open issues in future supplements to the licensing SER, NUREG-0847. In accordance with that commitment, this new section was introduced in SSER 5 and will be updated in subsequent SSERs. The current status of all CAPs and SPs follows. The status described here fully supersedes that described in previous SSERs.

1.13.1 Corrective Action Programs

(1) Cable Issues (TAC 71917)

Program review status:	NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), April 25, 1991 (the safety evaluation is reproduced in SSER 7 as Appendix P); review in progress.
Implementation status:	Full implementation expected by October 1992.
NRC inspections:	Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-22 (November 21, 1990); 50-390, 391/90-24 (December 17, 1990); 50-390, 391/90-27 (December 20, 1990); 50-390, 391/90-30 (February 25, 1991); 50-390, 391/91-07 (May 31, 1991); 50-390, 391/91-09 (July 15, 1991); 50-390, 391/91-12 (July 12, 1991); 50-390, 391/91-14 (August 22, 1991); to come.

(2) Cable Tray and Tray Supports (TAC R00516)

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 13, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3; review in progress.
Implementation status:	Full implementation expected by November 1993.
NRC inspections:	Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-22 (November 21, 1990); to come.

(3) Design Baseline and Verification Program (TAC 63594)

Program review status:	Complete: Inspection Report 50-390, 391/89-12 (November 20, 1989); NUREG-1232, Vol. 4.
Implementation status:	Full implementation expected by May 1992.
NRC inspections:	Inspection Reports 50-390, 391/89-12 (November 20, 1989); 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-20; (September 25, 1990); 50-390/91-201 (March 22, 1991); to come.

(4) <u>Electrical Conduit and Conduit Support (TAC R00508)</u>

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 1, 1989; NUREG-1232, Vol. 4; review in progress.
Implementation status:	Full implementation expected by August 1992.
NRC inspections:	Inspection Reports 50-390, 391/89-05 (May 25, 1989); 50-390, 391/89-07; (July 11, 1989); 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-20 (September 25, 1990); to come.

(5) <u>Electrical Issues (TAC 74502)</u>

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; review in progress.
Implementation status:	Full implementation expected by December 1992.
NRC inspections:	Inspection Reports 50-390, 391/90-30 (February 25, 1991); to come.

(6) <u>Equipment Seismic Qualification (TAC 71919)</u>

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; SSER 6, Sections 3.10.1 and 3.10.2; review in progress.
Implementation status:	Full implementation expected by May 1992.
NRC inspections:	Inspection Reports 50-390, 391/90-05 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-28 (January 11, 1991); 50-390, 391/91-03 (April 15, 1991); to come.

(7) <u>Fire Protection (TAC 63648)</u>

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 7, 1989; NUREG-1232, Vol. 4; review in progress, results to be published in Section 9.5.1 of a future SSER.
Implementation status:	Full implementation expected by May 1992.
NRC inspections:	To come.

(8) <u>Hanger and Analysis Update Program (TAC R00512)</u>

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 6, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3; review in progress.
Implementation status:	Full implementation expected by October 1992.

NRC inspections:	Inspection Reports 50-390, 391/89-14 (December 18, 1989); 50-390, 391/90-14 (August 3, 1990); 50-390, 391/90-18 (September 20, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-28 (January 11, 1991); 50-390, 391/91-03 (April 15, 1991); to come.

(9) Heat Code Traceability (TAC 71920)

Program review status:	Complete: Inspection Report 50-390, 391/89-09 (September 20, 1989); NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), March 29, 1991.
Implementation status:	Complete: Full implementation certified by letter, E. Wallace (TVA) to NRC, July 31, 1990; staff concurrence in SSER 7, Section 3.2.2.
NRC inspections:	Complete: Inspection Reports 50-390, 391/90-02 (March 15, 1990); 50-390, 391/89-09 (September 20, 1989).

(10) Heating, Ventilation, and Air-Conditioning Duct and Duct Supports (TAC R00510)

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 24, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3; review in progress.
Implementation status:	Full implementation expected by December 1992.
NRC inspections:	Inspection Reports 50-390, 391/90-05 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/91-01 (April 4, 1991); to come.

(11) Instrument Lines (TAC 71918)

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 8, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), October 26, 1990 (the safety evaluation was reproduced as Appendix K in SSER 6).
Implementation status:	Full implementation expected by November 1992.
NRC inspections:	Inspection Reports 50-390, 391/90-14 (August 3, 1990); 50-390, 391/90-23 (November 19, 1990); 50-390, 391/91-02 (March 6, 1991); 50-390, 391/91-03 (April 15, 1991); to come.

(12) Prestart Test Program (TAC 71924)

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), October 17, 1989; NUREG-1232, Vol. 4; letter P. S. Tam (NRC) to D. A. Nauman (TVA), March 27, 1991.

Implementation status:	TVA expects to complete and approve test results by September 1993.
NRC inspections:	Inspection Reports 50-390, 391/90-06 (April 25, 1990); 50-390, 391/90-12 (June 19, 1990); 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-14 (August 3, 1990); 50-390, 391/90-17 (August 14, 1990); 50-390, 391/90-20 (September 25, 1990); 50-390, 391/90-22 (November 21, 1990); 50-390, 391/90-24 (December 17, 1990); 50-390, 391/90-30 (February 25, 1991); 50-390, 391/90-33 (March 25, 1991); to come.

(13) Quality Assurance Records (TAC 71923)

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), December 8, 1989; NUREG-1232, Vol. 4; review in progress, results to be published in Section 17.3 of a future SSER.
Implementation status:	Full implementation expected by April 1993.
NRC inspections:	Inspection Reports 50-390, 391/90-06 (April 25, 1990); 50-390, 391/90-08 (September 13, 1990); 50-390, 391/91-08 (May 30, 1991); to come.

(14) Q-List (TAC 63590)

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 11, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), January 23, 1991.
Implementation status:	Full implementation expected by April 1992.
NRC inspections:	Inspection Reports 50-390, 391/90-08 (September 13, 1990); 50-390, 391/91-08 (May 30, 1991); to come.

(15) Replacement Items Program (TAC 71922)

Program review status:	Complete: Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), November 22, 1989; NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), February 11, 1991 (the safety evaluation was reproduced as Appendix N in SSER 6).
Implementation status:	Full implementation expected by April 1992.
NRC inspections:	Inspection Report 50-390, 391/91-08 (May 30, 1991); to come.

(16) Seismic Analysis (TAC R00514)

Program review status:	Letter, S. C. Black (NRC) to O. D. Kingsley (TVA), September 7, 1989; NUREG-1232, Vol. 4; SSER 6, Section 3.7; review in progress.

Implementation status:	Full implementation expected by January 1993.
NRC inspections:	Inspection Reports 50-390, 391/89-21 (May 10, 1990); 50-390, 391/90-20 (September 25, 1990); to come.

(17) Vendor Information Program (TAC 71921)

Program review status:	Complete: Letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), September 11, 1990 (the safety evaluation was reproduced as Appendix I in SSER 5).
Implementation status:	Full implementation expected by June 1993.
NRC inspections:	Inspection Report 50-390, 391/91-08 (May 30, 1991); to come.

(18) Welding (TAC 72106)

Program review status:	Complete: Inspection Reports 50-390, 391/89-04 (August 9, 1989); 50-390, 391/90-04 (May 17, 1990); NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), March 5, 1991.
Implementation status:	Full implementation expected by March 1993.
NRC inspections:	Inspection Reports 50-390, 391/89-04 (August 9, 1989); 50-390, 391/90-04 (May 17, 1990); 50-390, 391/90-20 (September 25, 1990); to come.

1.13.2 Special Programs

(1) Concrete Quality (TAC 63596)

Program review status:	Complete: NUREG-1232, Vol. 4.
Implementation status:	Complete: Full implementation certified by letter, E. Wallace (TVA) to NRC, August 31, 1990; staff concurrence in SSER 7, Section 3.8.2.1.
NRC inspections:	Complete: NUREG-1232, Vol. 4; Inspection Reports 50-390, 391/89-200 (December 12, 1989); 50-390, 391/90-26 (January 8, 1991)

(2) Containment Cooling (TAC 77284)

Program review status:	Complete: NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to D. A. Nauman (TVA), May 21, 1991 (the safety evaluation is reproduced as Section 6.2.2 of SSER 7).
Implementation status:	Full implementation expected by March 1992.
NRC inspections:	To come.

(3) Detailed Control Room Design Review (TAC 63655)

Program review status: Complete: NUREG-1232, Vol. 4; Section 18.1 and Appendix L of SSER 6.

Implementation status: Full implementation expected by October 1992.

NRC inspections: To come.

(4) Environmental Qualification Program (TAC 63591)

Program review status: NUREG-1232, Vol. 4; review in progress, results will be published in Section 3.11 of a future SSER.

Implementation status: Full implementation expected by July 1992.

NRC inspections: To come.

(5) Master Fuse List (TAC 76973)

Program review status: NUREG-1232, Vol. 4; letter, P. S. Tam (NRC) to O. D. Kingsley (TVA), February 6, 1991; review in progress.

Implementation status: Full implementation expected by April 1992.

NRC inspections: To come.

(6) Mechanical Equipment Qualification (TAC 76974)

Program review status: NUREG-1232, Vol. 4; review in progress, results will be published in Section 3.11 of a future SSER.

Implementation status: Full implementation expected by February 1992.

NRC inspections: To come.

(7) Microbiologically Induced Corrosion (TAC 63650)

Program review status: NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by May 1992.

NRC inspections: Inspection Reports 50-390, 391/90-09 (June 22, 1990); 50-390, 391/90-13 (August 2, 1990); to come.

(8) Moderate Energy Line Break Flooding (TAC 63595)

Program review status: NUREG-1232, Vol. 4; review in progress.

Implementation status: Full implementation expected by October 1992.

NRC inspections: To come.

(9) Radiation Monitoring Program (TAC 76975)

Program review status:	Complete: NUREG-1232, Vol. 4; this program covers areas addressed in Section 12 of the SER and SSERs.
Implementation status:	Full implementation expected by May 1993.
NRC inspections:	To come.

(10) Soil Liquefaction (TAC 77548)

Program review status:	NUREG-1232, Vol. 4; review in progress, results will be published in Section 2.5 of a future SSER.
Implementation status:	Full implementation expected by March 1992.
NRC inspections:	Inspection Reports 50-390, 391/89-21 (May 10, 1990); 50-390, 391/89-23 (February 21, 1990); to come.

(11) Use-as-Is CAQs (TAC 77549)

Program review status:	Complete: NUREG-1232, Vol. 4.
Implementation status:	Full implementation expected by July 1993.
NRC inspections:	Inspection Reports 50-390, 391/90-19 (October 15, 1990); 50-390, 391/91-08 (May 30, 1991); to come.

1.14 Implementation of Applicable Bulletin and Generic Letter Requirements

In SSER 5, Section 1.1, the staff stated that from time to time generic requirements or recommendations are issued in the form of bulletins and generic letters. The staff committed to prepare a summary of the implementation status of the applicable ones in SSER 6. The interim result of such effort was shown in Sections 1.14.1 and 1.14.2 of SSER 6. Because a long time has elapsed since these were addressed, the staff will reevaluate all bulletins and generic letters to determine if additional actions need to be taken. The staff will especially evaluate the appropriateness of implementation schedules. The evaluations will be completed before issuance of an operating license, and will be reported in a future SSER.

3 DESIGN CRITERIA--STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.2 Classification of Structures, Systems, and Components

3.2.2 System Quality Group Classification

3.2.2.5 Heat Code Traceability Corrective Action Program

In NUREG-1232, Volume 4, "Safety Evaluation Report on Tennessee Valley Authority: Watts Bar Nuclear Performance Plan," the staff stated that it will report the acceptability of TVA's implementation of the corrective action programs (CAPs) in the SSERs.

By letter dated July 31, 1990, TVA informed the staff that it has completed the Heat Code Traceability CAP, thus providing assurance that Watts Bar Unit 1 meets its licensing requirements concerning traceability of American Society of Mechanical Engineers (ASME) Code piping and pipe attachment materials.

The staff reviewed the programmatic aspects of the Heat Code Traceability CAP during a team inspection and found the CAP acceptable. The staff's review findings are documented in Inspection Report 50-390, 391/89-09, and in NUREG-1232, Volume 4. The staff also reviewed TVA's implementation in an additional team inspection and found that TVA had properly implemented this CAP. The inspection findings are documented in Inspection Report 50-390, 391/90-02 (March 15, 1990). The staff notes that issues related to this CAP that were uncovered by implementation of other CAPs will be resolved under the scope of those CAPs.

On the basis of its reviews and inspections, the staff concurs with TVA that the Heat Code Traceability CAP has been acceptably implemented for Unit 1.

3.7 Seismic Design

3.7.3 Seismic Subsystem Analysis

Surveys of earthquake damage (NUREG/CR-4776) have repeatedly pointed out the damage susceptibility of large, above-ground, vertical tanks under earthquake loads. The basic cause of damage has been identified as the inadequacy of the seismic analysis methods used for design of the tanks. The earlier commonly used method of analyzing tanks for seismic response was based on the Housner method, described in TID-7024, "Nuclear Reactors and Earthquakes," dated August 1963.

During the discussions related to the resolution of Unresolved Safety Issue (USI) A-40, "Seismic Design Criteria," the method of analysis of above-ground, flexible, vertical tanks was identified as an important topic requiring technical resolution. This USI is resolved in Revision 2 of Standard Review Plan (SRP) Sections 2.5.2, 3.7.2, and 3.7.3. The guidelines related to the seismic analysis of the above-ground vertical tanks are included in SRP Section 3.7.3.II.14.

Thus, a number of tanks at nuclear power plant sites are required to have confirmatory checks to ensure that the safety-related above-ground vertical tanks are adequately designed.

In order to confirm the design adequacy (i.e., the consideration of flexible tank wall) of the refueling water storage tank (RWST), the only safety-related above-ground vertical steel tank in the plant, the staff sent a request for additional information, dated June 27, 1989, to TVA. On October 19, 1990, TVA sent its response. The following evaluation was transmitted to the applicant by letter, P. S. Tam (NRC) to D. A. Nauman, March 28, 1991.

The RWST is a stainless steel, cylindrical, thin-walled shell structure with an outside diameter of 43.5 feet, supported on a circular, reinforced-concrete, slab foundation that is 3.7 feet thick. The foundation basemat is located at an elevation of 725.5 feet; the top of the tank roof is at an elevation of 773.5 feet. The roof of the tank is a spherical shell with a radius of 43.5 feet. To prevent roof damage from the possible effect of sloshing water during a seismic event, a freeboard distance of 4 feet is provided. This requirement leads to a maximum allowable water level of 34 feet in the tank. The tank wall thickness varies from 0.656 inch at the base to 0.3125 inch at the upper ring; the dome is 0.375 inch thick. The entire RWST structure including the base slab is founded on top of approximately 12 feet of crushed-stone fill, overlying a 7-foot-thick layer of basal gravel and a 13-foot-thick layer of weathered shale atop the bedrock. The basemat is embedded in a 3.7-foot-thick surface layer of Class A backfill material.

This tank was originally analyzed and designed for 0.18g modified Newmark ground-response spectrum based on the Housner method, as described in TID-7024. Under the Seismic Analysis Corrective Action Program (Seismic CAP) currently conducted by TVA, the tank was reanalyzed and evaluated for the site-specific ground-response spectrum (zero-period acceleration equals 0.215g) with the criteria documented in the Standard Review Plan (SRP), Revision 2 (NUREG-0800). The use of the site-specific spectrum for validating the original design structural features is acceptable for Watts Bar (see SER). To consider the soil-structure interaction (SSI) effects, the SASSI computer code was used for determining the seismic responses of the tank.

During the weeks of November 13-17 and December 18-22, 1989, and August 6-9, 1990, the staff and its consultants conducted an onsite inspection (see Inspection Report 50-390, 391/89-21) and a site audit (see publicly available memorandum, P. S. Tam to Document Control Desk, October 19, 1990), respectively, of the Seismic CAP plan and the implementation of the CAP plan. The staff concluded that the modeling techniques of the RWST superstructure and soil foundation, the procedures for generating the amplified response spectra, and the method for evaluating the dynamic stability (overturning and sliding) met the SRP guidance, and the final seismic responses (axial forces, shear forces, bending moments, base shear, and overturning moment) are acceptable. Therefore, this issue of wall flexibility of vertical steel tanks at Watts Bar is resolved. However, because of the incomplete status of the design calculations, the evaluation of the RWST structural integrity (bucking of tank wall, nozzle integrity, anchorages, resistance of hoop tension, integrity of tank roof, etc.) has not been reviewed by the staff. According to TVA, these calculations would not be completed until the end of June 1991. The staff has now reviewed TVA's tank calculations during a site audit (September 9-13, 1991), and will report findings in an audit report associated with the Seismic Analysis CAP (see Section 1.13.1).

On the basis of the review findings discussed above, the staff concludes that the criteria used by TVA for tank evaluation met the SRP guidance and, therefore, the issue of wall flexibility of vertical steel tanks for Watts Bar is considered resolved. This effort was tracked by TAC 73097 and 73098.

3.8 Design of Seismic Category I Structures

3.8.2 Concrete and Structural Steel Internal Structures

3.8.2.1 Special Program on Concrete Quality

In NUREG-1232, Volume 4, "Safety Evaluation Report on Tennessee Valley Authority: Watts Bar Nuclear Performance Plan," the staff stated that it will report the acceptability of TVA's implementation of the special programs (SPs) in SSERs. When NUREG-1232, Volume 4, was published, the SP on concrete quality was already at an advanced stage of completion.

By letter dated August 31, 1990, TVA informed the staff that it has completed the SP on concrete quality.

The staff inspected implementation of this SP, and has documented its findings in Inspection Reports 50-390, 391/89-200 (December 12, 1989), and 50-390, 391/90-26 (January 8, 1991). The staff found that TVA has properly implemented this SP.

On the basis of the above review and inspections, the staff concurs with TVA that the Concrete Quality Special Program has been acceptably implemented for Unit 1.

3.9 Mechanical Systems and Components

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.3 Design and Installation of Pressure-Relief Devices

In SSER 6, the staff identified Outstanding Issue 19(i) regarding the design and installation of the main steam safety valves (MSSVs). In a letter dated May 8, 1991, the staff specifically identified issues regarding the adequacy of the MSSV header, discharge piping, and supports to withstand the loads resulting from valve discharge and other appropriate loads. The staff also identified issues regarding the adequacy of the MSSVs to achieve full capacity and acceptable blowdown. The following evaluation was sent to the applicant by letter, P. S. Tam to D. A. Nauman (TVA), July 31, 1991.

> In a letter dated June 21, 1991, the applicant responded to the staff's concern regarding the adequacy of the MSSV design and installation. The applicant stated that all valve and piping components have been analyzed for all MSSV discharge loads acting simultaneously combined with other required loads in accordance with Table I of SRP Section 3.9.3, which is acceptable to the staff. The applicant also

stated that although the individual plant MSSVs were not tested, full-pressure, full-flow tests of a large population of representative Dresser model valves were performed by Dresser Industries. Results were then used to establish the MSSV adjustment ring settings for the Watts Bar plant valves. These settings ensure full-rated steam capacity and limit the blowdown to less than 10 percent. The applicant stated that the maximum acceptable blowdown is 10 percent for meeting the plant design basis. The staff agrees that the applicant has demonstrated the adequacy of the MSSV design and installation; therefore, this resolves Outstanding Issue 19(i).

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.1 Containment Structure

Maximum Pressure and Temperature Analysis

The following evaluation was issued by letter, D. E. LaBarge and P. S. Tam (NRC) to D. A. Nauman (TVA), April 24, 1991, under both Sequoyah and Watts Bar dockets.

> In the safety evaluation report (letter, C. O. Thomas (NRC) to E. P. Rahe (Westinghouse), August 22, 1983) of Westinghouse Topical Reports WCAP-8821 and WCAP-8822, regarding mass and energy releases following a main steamline break (MSLB), the staff identified a concern of possible elevated temperatures in containments that could affect the equipment qualification in ice-condenser containments. Following an MSLB, the uncovered tube bundles would allow heat to be transferred to steam in the steam generator, which would result in the release of superheated steam to the containment. The effects of the superheated steam release are more pronounced in a small area, such as the lower compartment of an ice-condenser containment. The containment temperatures could be much higher than those previously calculated in the final safety analysis reports (FSARs) of ice-condenser plants. In the FSARs, a saturated steam release was assumed for MSLBs. The previously calculated containment temperature profiles (with a maximum of 327°F) used for equipment qualification inside ice-condenser containment might no longer be valid, and the maximum temperature could be exceeded.
>
> As part of its review for the Catawba facility, the staff identified the concern of the MSLB inside ice-condenser containments as proposed License Condition 17. This proposed license condition required additional information on testing and analysis to justify the adequacy of equipment qualification following an MSLB inside containment with superheated steam release. By letter dated November 27, 1985, Westinghouse submitted Topical Reports WCAP-10986P/10987 and WCAP-10988P/10989, which contained the required information specified by the license condition. Westinghouse, using the proposed drain-flow heat-transfer model, revised the LOTIC-III computer code, and reanalyzed MSLBs inside ice-condenser containments including the effects of superheated steam released. The results showed that the bulk-average containment temperature was below the equipment qualification temperature of 327°F. On the basis of the information in the topical reports, the staff concluded that the Catawba license condition had been resolved. However, the staff was continuing its review to confirm the validity and accuracy of the models and assumptions used in the revised MSLB

analyses. Because of the similarity of design, this concern also applies to McGuire, Sequoyah, and Watts Bar plants.

In the safety evaluation report [letter, G. Holahan (NRC) to W. J. Johnson (Westinghouse), April 1, 1991], the staff found Topical Report WCAP-10986P acceptable. The calculated global containment temperature following a superheated steam release is within the qualification temperature of 327°F. However, the staff was concerned that the local temperature near the superheated steam jet may exceed the value of 327°F. The staff decided that the issue of local temperatures would be addressed in separate plant-specific evaluations. In response to the staff's concern about the local high temperatures, the applicant, by letters dated August 17 and November 3, 1989, assessed the elevated temperature effects in the vicinity of the breaks in Sequoyah and Watts Bar containments. The staff's evaluation of these submittals is in the following paragraphs.

In WCAP-10988P, Westinghouse performed a containment analysis for an MSLB with superheated steam release using the three-dimensional multinode code, COBRA-NC. Under an NRC contract, Argonne National Laboratory (ANL) performed an independent confirmatory analysis [letter, W. T. Sha (ANL) to B. L. O. Grenier (NRC), March 31, 1989] using the COMMIX code. The results from both analyses confirmed that the global containment temperature was lower than the equipment qualification temperature of 327°F and that the results from the LOTIC-III code were conservative. However, the results of both analyses showed that local temperatures near the superheated steam jet exceeded 327°F. The elevated temperature effect near the breaks was more pronounced in the COMMIX results. In its August 17 and November 3, 1989, letters, TVA responded to the staff's concern by comparing the COBRA-NC and COMMIX analyses, evaluating all potential MSLB locations at Sequoyah and Watts Bar, and assessing the impact on equipment needed for safe shutdown.

One major difference between the two codes is that the COBRA-NC can model two-phase flow, but the COMMIX code is limited to single-phase flow. As a result, the COMMIX code cannot model the transport process of liquid droplets of water in a steam and noncondensable gas mixture. In the limiting case of the COMMIX analyses, ANL assumed the droplets being uniformly entrained in the mixture of steam and noncondensable gas. The heat sinks from liquid droplets were thus distributed accordingly. This assumption is conservative for calculating global temperature, but it is not realistic for calculating local temperature. In reality, the liquid droplets should be distributed with higher concentration in the vicinity of the break jet. The COBRA-NC analysis, which is provided with two-phase capability, can calculate the distribution of liquid droplets instead of assuming a limiting distribution. Therefore, the COBRA-NC code obtained more realistic best-estimate local temperatures than the COMMIX code.

Another difference identified by ANL in the COMMIX modeling was in the assumed boundary condition relative to the postulated break location. In the COMMIX model, the break location was conservatively assumed at the boundary of the cell, that is, at the surface of the crane wall.

In the COBRA-NC model, the break was postulated at the center of the cell (approximately 4 feet from the crane wall) so that the flow could be mixed more effectively. In its November 3, 1989, letter, TVA evaluated the steamlines in the containments of the Sequoyah and Watts Bar plants for all potential break locations. The postulated break at Sequoyah and Watts Bar plants is located approximately 1/4 distance between the crane wall and the biological shield wall, which is more than 4 feet from the crane wall. Therefore, the break location assumed in the COBRA-NC code was more realistic for Sequoyah and Watts Bar plants.

On the basis of this comparison, TVA determined that the temperatures calculated by the COBRA-NC code in the vicinity of breaks were more realistic than the results calculated by the COMMIX code. Therefore, TVA used the local temperatures predicted by the COBRA-NC analysis for the assessment of the adequacy of equipment qualification.

A plant-specific COBRA-NC analysis has not been performed for Watts Bar and Sequoyah. The hot-spot temperature is based on the Catawba Nuclear Station model given in WCAP-10988P. The main primary system components at Catawba, Watts Bar, and Sequoyah are in the same relative locations. However, the ice-condenser drain locations at Watts Bar and Sequoyah are grouped closer together around the steam generator and main steamlines. Unlike Catawba, the Watts Bar and Sequoyah ice-condenser drains would flow directly into the break node. Therefore, the superheated vapor from the break would be immediately brought into contact with a large amount of subcooled water that would result in even lower break node temperatures for the Watts Bar and Sequoyah plants.

Furthermore, TVA reviewed all equipment needed for plant shutdown following an MSLB inside the containment for its proximity to any steamline. None of this equipment is located in the hot-spot region predicted by COBRA-NC. Therefore, the qualification and performance of the equipment will not be impaired by the elevated temperature near the breaks.

On the basis of this plant-specific evaluation, the staff concludes that the local temperatures calculated by the COBRA-NC code are realistic and acceptable for equipment qualification evaluation. In the Sequoyah and Watts Bar containments, the equipment needed for plant shutdown following an MSLB is sufficiently far away from the hot spot of breaks. Therefore, the staff's concern regarding local temperatures near MSLBs inside the Sequoyah and Watts Bar containments is resolved. In conjunction with the SER for Topical Reports WCAP-10986P and WCAP-10988P, the staff concludes that the concern over the superheated steam released in the containments of Sequoyah and Watts Bar plants is resolved. This effort was tracked by TAC 63621.

6.2.2 Containment Heat Removal Systems

The following safety evaluation was issued by letter, P. S. Tam (NRC) to D. A. Nauman (TVA), May 21, 1991.

In Section 3.3.2 of the Safety Evaluation Report (NUREG-1232, Volume 4) regarding the Watts Bar Unit 1 Nuclear Performance Plan, the staff required TVA to determine the mode in which the plant will be maintained following a main steamline break (MSLB). This issue is part of the special program on containment cooling. By letters dated August 25, 1989 (revised final report on CAQR WBF 870061), and July 31, 1990, TVA clarified that hot standby was the design-basis safe-shutdown mode following an MSLB and identified resulting design changes to the containment cooling system. Because hot standby is the safe-shutdown condition, the reactor coolant system may remain at elevated temperatures for an extended period of time following an accident, including an MSLB. This extended time at elevated temperature had not been considered in the long-term containment temperature analyses performed for equipment qualification purposes. TVA, therefore, reanalyzed the containment temperature profiles to account for the extended time at hot standby.

As a result of this reanalysis, the lower containment cooling system had to be modified to allow long-term safety-grade cooling of the lower containment in conjunction with containment spray and ice-condenser systems. These modifications include the installation of new safety-grade fans, fan motors, backdraft dampers, and cabling, in addition to upgrading the existing ductwork and housings to safety-grade requirements. TVA committed to complete those modifications before loading fuel into each unit.

The staff has reviewed the assumptions used in TVA's reanalysis of the containment temperature over the extended time period and concludes that they are conservative. On the basis of this reanalysis and the lower containment cooling system modifications, the staff concludes that hot standby is an acceptable mode following a main steamline break and that the containment cooling system modifications are acceptable. Final temperature profiles and environmental qualification of equipment will be evaluated as part of the staff's evaluation of the environmental qualification program.

6.3 Emergency Core Cooling System

6.3.1 System Design

By letter dated September 17, 1986, the applicant proposed a design change to remove from Unit 1 the upper head injection (UHI) system. By letter dated September 19, 1985, the applicant informed the staff of its intention not to install the UHI system at Unit 2. The UHI system was originally designed for use in Watts Bar to enhance core cooling during the blowdown phase of the loss-of-coolant accident (LOCA). The UHI system is being eliminated to increase operational flexibility. The staff previously approved a similar design change for the McGuire plant. The staff has reviewed the request for the design change and the supporting analytical results documented in the FSAR up to Amendment No. 63 and issued the following evaluation to the applicant by letter, P. S. Tam (NRC) to D. A. Nauman (TVA), March 29, 1991.

6.3.1.1 Deletion of the Upper Head Injection System

The applicant used the approved LOFTRAN computer model to reanalyze the following two transients: (1) the inadvertent opening of a steam generator relief or safety valve (Table 15.2-1 of the FSAR) and (2) the steamline break (Table 15.4-1 of the FSAR). These two transients were analyzed because these were the only two transients that were predicted to depressurize the primary system sufficiently to actuate the UHI system. The applicant performed an analysis to determine whether departure from nucleate boiling (DNB) would occur for both transients, and results confirmed that no DNB would occur. This ensures that no fuel failure would result from the transients. The staff concludes that the applicant's transient analysis is adequate and acceptable since an approved method was used. The results were found to be acceptable since specific acceptable fuel design limits were not exceeded.

The applicant performed the small-break LOCA analysis using approved methods, that is (1) the NOTRUMP code (Westinghouse Topical Reports WCAP-10080 and WCAP-10081) for the calculation of transient depressurization of the reactor system, core power, water-steam mixture height, and steam flow past the uncovered portion of the core and (2) the LOCTA code (Westinghouse Topical Report WCAP-8305) for the peak cladding temperature analysis. This analysis was done assuming 102 percent of the core power of 3411 MWt and a total peaking factor of 2.40. Various break sizes were analyzed and results showed that the worst break size is a 4-inch-diameter break that results in the highest peak cladding temperature of 1549°F, well below the accepted limit of 2200°F. The staff concludes that the applicant's small-break LOCA analysis is acceptable since approved methods were used to show the analytical results to be within the acceptance criteria specified in 10 CFR 50.46.

The applicant performed a large-break LOCA analysis supporting its request for removal of the UHI system. In the applicant's submittal, only double-ended, cold-leg, guillotine (DECLG) breaks were analyzed since they were identified previously as limiting cases that result in the highest peak cladding temperature. The DECLG break analysis was performed by assuming a peaking factor of 2.40, 102 percent of the core power of 3411 MWt, and a loss of offsite power at the beginning of the accident. A sensitivity study of DECLG break sizes on the effect of the peak cladding temperature was performed by using Moody discharge coefficients of 0.4, 0.6, and 0.8. The results showed that that the DECLG with a discharge coefficient of 0.6 is the worst large-break LOCA case, resulting in a peak cladding temperature of 2193°F. The analysis was performed by using a modified revision of the 1981 Westinghouse emergency core cooling system (ECCS) evaluation model (WCAP-9220-P-A, Rev. 1). This evaluation model used the revised PAD fuel thermal safety model (WCAP-8720) for calculating the intial fuel conditions; the SATAN-VI code (WCAP-8302) for the thermal hydraulic calculation during the blowdown period; the transient WREFLOOD (WCAP-8170) and BASH (WCAP-10266 and addendum) codes for calculating the refill and reflood transient periods; the LOCBART code (WCAP-8305) for calculating the peak cladding temperature; and the LOTIC code (WCAP-8355) for calculating the ice-condenser containment pressure transient. The staff found that the approved analytical methods and computer codes were used, and the results showed that peak cladding temperature, metal-water reaction, and cladding oxidation are within the acceptance criteria specified in 10 CFR 50.46 for the LOCA analysis.

On the basis of this evaluation, the staff concludes that it is acceptable to modify the original Watts Bar ECCS design (as described in the Watts Bar SER, NUREG-0847) to delete the UHI system from both units. This review was tracked by TAC 77195.

7 INSTRUMENTATION AND CONTROLS

7.4 Systems Required for Safe Shutdown

7.4.2 Shutdown From Auxiliary Control Room

By letter dated September 26, 1985, the applicant requested a deviation from the guidelines of Standard Review Plan (SRP) Section 9.5.1 Subsection 2*, regarding the installation of reactor coolant system (RCS) cold-leg temperature (T-cold) instrumentation in the auxiliary control room (ACR). The staff's evaluation of the applicant's request was issued by letter, P. S. Tam (NRC) to D. A. Nauman (TVA), May 17, 1991. That safety evaluation follows.

> SRP Section 9.5.1, Subsection 2, Item B, states that "The process monitoring function should be capable of providing direct readings of the process variables necessary to perform and control the above functions." Specifically, this means the alternative or dedicated shutdown capability provided for a specific fire area shall be able to (1) achieve and maintain subcritical reactivity conditions in the reactor, (2) maintain reactor coolant inventory, (3) achieve and maintain hot-standby conditions, (4) achieve cold shutdown within 72 hours, and (5) maintain cold-shutdown conditions thereafter. During a post-fire shutdown, the reactor coolant system process variables shall be maintained within those predicted for a loss of normal ac power, and the fission product boundary integrity shall not be affected; that is, there shall be no fuel cladding damage, rupture of any primary coolant boundary, or rupture of the containment boundary. Reactor coolant system cold-leg temperature indication is one of the process variables typically provided to aid in assessing the establishment of natural circulation cooling.
>
> In its September 26, 1985, submittal, the applicant justified not installing T-cold instrumentation in the ACR. The applicant proposed to use T-sat (saturation temperature corresponding to steam generator pressure) instead of T-cold instrumentation in the ACR. The applicant has stated that indications of RCS subcooling, T-hot stable or decreasing, and steam generator pressure stable or decreasing are available in the ACR to indicate loss of natural circulation cooling. Furthermore, all of these indications are specified for use in the Watts Bar emergency procedures to verify adequate natural circulation, and the operators are periodically trained to shut the plant down from the ACR.
>
> The instrumentation in the Watts Bar ACR for steam generator pressure indication will be enhanced by a dual scale to read saturation temperature and thus allow the operator to infer T-cold. On the basis of

*The applicant indicated that deviation is from 10 CFR Part 50, Appendix R, Section III.L.2.d. The requested deviation actually pertains to the quoted SRP section.

data obtained during startup testing at the Sequoyah Nuclear Station and Diablo Canyon plant, the applicant provided the results of a statistical evaluation of the relationship between T-sat and T-cold startup test measurements to demonstrate the accuracy of inferring T-cold from T-sat. In terms of T-cold minus T-sat, the following results were obtained:

Temperature	Sequoyah	Diablo Canyon
Mean	4.33°F	4.67°F
Standard deviation	3.29°F	1.65°F

The temperature differences noted above are within the tolerance and accuracy levels of the instrumentation. In obtaining the data, the Sequoyah cooldown was terminated at approximately 465°F, whereas the Diablo Canyon cooldown was continued to below residual heat removal system initiation. Since the data (T-cold and steam generator pressure) were obtained simultaneously during the cooldown tests, the data demonstrated the adequacy of using steam generator pressure to determine T-sat and infer T-cold, as well as the lack of significant time lag between the two indications. On the basis of the similarity of the Watts Bar and Sequoyah plant design and process variable instrumentation associated with the natural circulation cooldown process, the referenced test data are considered applicable to Watts Bar.

The applicant has stated that the natural circulation test at Sequoyah was performed from the main control room and the equipment (not controls) used during the natural circulation test is identical to the equipment that would be used in an Appendix R shutdown from the ACR for similar plants, such as Watts Bar. For example, the auxiliary feedwater pumps, the centrifugal charging pumps, essential raw cooling water pumps, and component cooling system pumps are used for natural circulation cooldown from the ACR and were also used during the main control room test. Therefore, the test results are applicable to a shutdown from the ACR. Also, the instrumentation provided for verification of natural circulation is consistent with the Westinghouse Owners Group emergency response guidelines.

On the basis of the natural circulation tests at Sequoyah and Diablo Canyon, the staff finds that T-sat and T-cold trend together reasonably well; furthermore, the Watts Bar operators have been trained in the use of steam generator pressure. Also, the applicant has the ability to monitor RCS subcooling and to monitor all four steam generators from the ACR. Therefore, the staff concludes that the use of T-sat instead of T-cold in assessing natural circulation cooling in the RCS is an acceptable deviation from the guideline of SRP Section 9.5.1.

The staff's review effort was tracked by TAC 63607.

7.7 Control Systems Not Required for Safety

7.7.2 Bypassed and Inoperable Status Indication System*

In the SER, the staff stated that the applicant should address guidelines of Regulatory Guide (RG) 1.47, Revision 0, "Bypassed and Inoperable Status Indications for Nuclear Power Plant Safety Systems." This item was identified as proposed License Condition 10.

In a letter dated January 29, 1987, the applicant submitted a document entitled, "Functional Requirements Document for the Bypassed and Inoperable Status Indication (BISI) System" for the proposed system at Watts Bar. The staff reviewed this letter and concluded that it needed additional information in order to complete its evaluation. This request for additional information was submitted to TVA on August 13, 1990. TVA responded to this request by letter dated October 22, 1990. The staff also reviewed FSAR Section 7.7.1.3.6 as revised by Amendment No. 63.

The BISI system document defines the required functional and operational characteristics for the BISI to meet the guidelines of RG 1.47. Each unit has a separate BISI system. The operating and trip bypass of the reactor protection system (RPS) and the instrument and logic portion of the engineered safety features actuation system (ESFAS) are not included in the BISI system. The RPS and ESFAS bypass requirements are given in Section 4.13 of the Institute of Electrical and Electronics Engineers (IEEE) Standard 279-1971, which is endorsed by 10 CFR 50.55a, "Codes and Standards."

IEEE Standard 279-1971 and Criterion XIV of Appendix B, 10 CFR Part 50, require that systems actuated or controlled by the protection system perform their intended functions. This includes those auxiliary or supporting systems that must be operable in order for the protection system and the system it actuates to perform their intended functions. The BISI system provides automatic main control room (MCR) indication if the protective action of some part of the protection system has been bypassed or deliberately rendered inoperable for any purpose, including periodic tests or maintenance.

The BISI system is designed to meet the following guidelines:

- The bypassed or inoperable condition affects a system that is designed to perform automatically a function that is important to public safety.

- The bypass will be utilized by plant personnel if the inoperable condition can be reasonably expected to occur more frequently than once per year.

- The bypass or inoperable condition is expected to occur when the system is normally required to be operable as required by the technical specifications.

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*In the SER, this section was entitled "Safety System Status Monitoring System." The title change reflects a name change of the system. The contents of the section were issued as an enclosure to a letter, P. S. Tam (NRC) to D. A. Nauman (TVA), dated March 28, 1991.

The applicant identified the following systems to be monitored and alarmed on a system level by the BISI system:

- main and auxiliary feedwater (including steam generator isolation)
- safety injection
- residual heat removal
- containment building spray
- emergency gas treatment
- essential raw cooling water
- chemical and volume control
- ventilation
- component cooling water
- control air (including auxiliary control air)
- standby diesel generator

Portions of these systems that are not safety related and can be separated from the safety function performed by these systems will not be monitored. Components and systems required to support these safety-related systems will be monitored by the BISI system.

The BISI system is designed to supplement administrative procedures during normal plant operating conditions, with automatic indication of the bypassed or inoperable status of each redundant portion of a system that performs a function important to safety. Manual capability has been provided to activate the safety-related system BISI indicators for systems that are in an inoperable or bypassed condition, whether deliberately or otherwise induced, which are not automatically indicated. In addition to the BISI upper-level indication, an audible alarm will alert operators in the MCR when a BISI system alarms. A BISI system alarm can be printed on demand, by shift turnover, or for historical logging.

The applicant was asked to identify any systems or components that were not being monitored because they were not expected to be rendered inoperable more than once a year. In its October 22, 1990, response, the applicant stated that it did not exclude any components because they were not expected to be rendered inoperable more than once a year.

The BISI system is not required to operate during or after an accident, nor is it designed to safety system criteria; however, the design should allow testing during normal operations. FSAR Chapter 14, Table 14.2, identifies a preoperational test (No. TVA-64) to demonstrate the ability of the Technical Support Center (TSC) to acquire and process non-safety-related data from throughout the plant via multiplexer inputs for display and alarming in the main control room and the TSC.

In Section 3.5.3, "Component Level Implementation [BISI] Criteria," in design document WB-DC-30-8, the applicant indicated why the main steam isolation system was not included in the systems identified in Section 6.1. The exclusion is based on the criterion that components that fail in the safe direction upon loss of power would not be monitored. This is acceptable.

Electrical isolation between the safety system inputs and the non-safety-graded BISI are the same as for the safety parameter display system (SPDS), as described in the applicant's letter of November 1, 1990. The staff, in its review of the Watts Bar SPDS regarding isolation from electrical and electronic interference with equipment and sensors, concluded that the applicant has satisfied the NUREG-0737, Supplement 1 requirement (see SSER 6, Section 18.2). TVA's method of isolation is to provide a Potter & Brumfield relay model KUIP-3A11-120 or KUIP-3A11-130 for 120-V ac or 125-V dc application. The isolation is between the coil (safety system) and the contacts (non-safety-graded BISI system). The staff concurs that the applicant has provided adequate isolation between the safety systems and the BISI system.

The staff reviewed the following schematic diagrams and BISI design information from Watts Bar's System Requirement Specification (SRS). The staff's review of the schematic diagrams indicates that the data in the BISI system agree with design criteria.

Item	Schematic diagram	SRS
Motor-driven auxiliary feedwater pump 1A-A	1-45W760-3-1, Rev. 1, Design Control Authorization (DCA) DCA-P03365-60,61,62	4.4.37.4
Valve FCV-3-33-A	1-45W760-3-6, Rev. 0, DCA-P03365-97,98,99	
Control bldg. emergency pressurization fan A-A	1-45W760-31-10, Rev. 0, DCA-P03223-4,5	4.4.37.8
Damper FCO-31-6	1-45W600-30-5, Rev. 0	
Safety injection pump A-A	1-45W760-63-1, Rev. 0, DCA-P03225-11, 12,13,14,15; DCA-P03427-204,207	4.4.37.14
Valve 1-FCV-63-26-A	1-45W760-63-3, Rev. 0, DCA-P03225-34, 35,36,37,38,39,40,41,42	

The staff concludes that the BISI design at Watts Bar conforms with the requirements of

- Criterion XIV, "Inspection, Test, and Operating Status," of Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"
- 10 CFR 50.55a, "Codes and Standards," paragraph (h), given in the Institute of Electrical and Electronics Engineers, "Criteria for Nuclear Power Plant Protection Systems" (IEEE 279), Section 4.13
- Class 1E inputs into the non-Class 1E BISI system also comply with the guidelines of RG 1.75, "Physical and Electrical Independence for Nuclear Power Plants"
- RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plants," May 1973

On the basis that the BISI system meets all pertinent criteria, and that it will be fully implemented before fuel load, the staff considers proposed License Condition 10 fully resolved.

8 ELECTRIC POWER SYSTEMS

8.3 Onsite Power Systems

8.3.1 Onsite ac Power System Compliance With GDC 17

8.3.1.2 Low and/or Degraded Grid Voltage Condition

In the SER, the staff stated that it would verify the adequacy of the applicant's analysis regarding compliance with Branch Technical Position (BTP) PSB-1 once the preoperational test was completed.

The staff noted that the preoperational test has shown that the Watts Bar design conforms with BTP PSB-1 (see Inspection Report 50-390/84-90, dated February 11, 1985). The staff is still evaluating the status of this issue and will update the status in a future SSER.

8.3.1.6 Diesel Generator Reliability Qualification Testing

In SSER 2, the staff indicated that it would verify diesel generator qualification testing. The applicant gave the staff a replacement copy of the December 30, 1975, submittal, which transmitted the diesel generator qualification test report. On the basis of an audit review of this report, the staff verified that the Watts Bar diesel generators have been satisfactorily tested in accordance with the start and load acceptance qualification guidelines of Section 6.3.2 of Institute of Electrical and Electronics Engineers (IEEE) Standard 387-1977. This resolves Confirmatory Issue 29.

8.3.3 Common Electrical Features and Requirements

8.3.3.4 Compliance With NUREG-0737 Items

Emergency Power for Pressurizer Equipment (II.G.1)

To meet the guidelines of both NUREG-0737 Item II.G.1 and BTP RSB 5-2, the SER required, as a condition to the license, that power is supplied for the power-operated relief valves and block valves from the same power train but from different buses. By letter dated February 6, 1985, the applicant indicated its commitment to comply with this license condition before fuel loading. This eliminates the need for proposed License Condition 17.

8.3.3.6 Compliance With GDC 50

In the SER, the staff required a reevaluation of the penetrations' capability to withstand, without seal failure, the total range of available time-current characteristics assuming a single failure of any overcurrent protective device. In SSER 3, the staff found the results of the applicant's reevaluation acceptable pending confirmation that information presented in a January 17, 1984, letter was incorporated into the FSAR. On the basis of a review of information documented in the FSAR through Amendment 63, the staff has reconfirmed that the applicant's reevaluation is acceptable. Therefore, the staff considers information presented in the January 17, 1984, letter to be incorporated into the FSAR, and Confirmatory Issue 35 resolved.

14 INITIAL TEST PROGRAM

14.2 Test Program

Although no reason was specifically given in the SER for proposed License Condition 31 (due to an inadvertent editorial error), it was originally intended to require that TVA report to the staff within 30 days whenever an approved initial test is modified.

By letter dated July 1, 1991, the applicant provided a commitment stating:

> Any changes that are made under the provisions of 10 CFR 50.59 to the initial Startup Test Program, as described in Final Safety Analysis Report, Chapter 14, that normally would be reported to NRC annually in accordance with 10 CFR 50.59(b) will be reported within one month that such change is made effective.

The staff determined that this commitment is comparable to that made by other applicants, and that the commitment will provide the information required by proposed License Condition 31. This determination eliminates the need for proposed License Condition 31.

15 ACCIDENT ANALYSIS

15.2 Normal Operation and Anticipated Transients

15.2.4 Reactivity and Power Distribution Anomalies

15.2.4.1 Uncontrolled Rod Cluster Assembly Bank Withdrawal From Zero-Power Conditions

The original Watts Bar Final Safety Analysis Report (FSAR) analysis for the zero-power uncontrolled rod cluster control assembly (RCCA) bank withdrawal event assumed all four reactor coolant pumps to be running. However, the early version of the draft Technical Specifications (TS) would require only one reactor coolant pump to be running while in Modes 3 and 4, when a potential for RCCA withdrawal might exist, thus raising a question as to the adequacy of the analysis assumptions for this event in these modes.

Subsequently, the draft TS for Watts Bar were changed to increase the required number of operating pumps to two in Modes 3 and 4 when rod withdrawal is possible (trip system breakers are closed). In a letter from J. Domer to E. Adensam (NRC), dated June 21, 1985, the applicant submitted a new analysis for the event in these modes, with two pumps in operation. The methodology used for the analysis was standard Westinghouse methodology, commonly used for analysis of this event. The staff's initial review of the analysis concluded that it was acceptable. However, on January 10, 1986 (letter from J. Domer to B. Youngblood), TVA notified the NRC that the zero-power RCCA withdrawal analysis above had used the WBR-1 critical heat flux correlation. This correlation has been approved and used for other reactors, but for Watts Bar the W-3 R-Grid correlation is used to form the licensing basis. The change to this correlation in the two-pump, zero-power, rod-withdrawal analysis, as carried out in the analysis of June 21, 1985, would have resulted in exceeding the departure from nucleate boiling ratio (DNBR) safety limit. To solve this problem, Westinghouse has reanalyzed the event, using the W-3 R-Grid correlation and, in addition, improved analytical methodology. This change was submitted to the NRC in a May 14, 1986, letter from R. Gridley to B. Youngblood.

The initial Watts Bar zero-power RCCA withdrawal analysis used the Westinghouse WIT-6 code for the transient analysis. This has been commonly used for this event and has been accepted by the staff in numerous past reviews for other reactors. The core neutronic analysis in WIT-6 is by point kinetics, using externally developed, conservatively bounding reactivity coefficients, RCCA reactivity worths, and power distributions. The WIT-6 calculation is followed by FACTRAN and THINC calculations to determine heat flux and temperature transients, and to determine transient DNBR. These codes are also commonly used and have been accepted for these types of analyses to determine minimum DNBR.

The principal change to the new analysis (in addition to using the W-3 R-Grid correlation) was the use of TWINKLE rather than WIT-6 for the transient analysis. As a further addition, the "Power Range High Positive Neutron Flux Rate Trip" was added to the list of automatic protection system features.

The TWINKLE code uses the multidimensional (up to three dimensions) core neutronics rather than point kinetics to provide a better representation of the core, the reactivity feedback, RCCA reactivity changes, and power distributions. It has been used and approved for many years for many Westinghouse transient analyses, including the rod ejection accident, and has been used in recent years for the zero-power RCCA withdrawal analysis in FSARs (e.g., Seabrook) in the same manner as presented for Watts Bar. The TWINKLE analysis is (as with the WIT-6 analyses) followed by the FACTRAN and THINC thermal analyses. This code package has been previously reviewed and approved and is acceptable for use in the Watts Bar zero-power RCCA withdrawal analysis. The result of the analysis, that is, minimum DNBR, falls within the approved limits for the W-3 R-Grid correlation and is acceptable.

The applicant has revised pertinent sections of the Watts Bar FSAR by Amendments up to No. 66 to reflect the changes indicated above. This includes deletion of references to WIT-6 and to moderator reactivity coefficients (not needed in TWINKLE), and addition of references to TWINKLE and new curves presenting the results of the analyses. The staff has concluded that the revisions are acceptable. This review was tracked by TAC 63605.

APPENDIX A

CHRONOLOGY OF RADIOLOGICAL REVIEW OF WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2, OPERATING LICENSE REVIEW

NRC Letters, Memoranda, and Summaries

May 10, 1990	Letter from B. D. Liaw to O. D. Kingsley, Jr. (TVA) forwarding NRC Inspection Report 50-390, 391/89-21.
October 19, 1990	Memorandum from P. S. Tam to NRC Document Control Desk, regarding onsite audit on Watts Bar Seismic Analysis CAP plan.
January 4, 1991	Letter from P. S. Tam to O. D. Kingsley (TVA), forwarding safety evaluation regarding FSAR Amendments 54-64 and asking TVA to address open issues identified in the enclosures.
January 8, 1991	Letter from P. S. Tam to O. D. Kingsley (TVA), requesting additional information required for completion of review of proposed use of Eagle-21 system microprocessor to replace originally approved resistance temperature detectors (RTDs).
January 9, 1991	Letter from P. S. Tam to O. D. Kingsley (TVA), informing TVA that its response to Generic Letter 90-06 regarding resolution of Generic Issues 70 and 94 is acceptable.
January 10, 1991	Notice of meeting of January 29, 1991, with TVA to discuss lack of seismic Category I classification for cable trays and conduits.
January 14, 1991	Letter from B. A. Wilson to O. D. Kingsley (TVA), forwarding summary of December 12, 1990, meeting regarding plans to assess quality of required records.
January 22, 1991	Notice of licensing action status meeting on January 31, 1991.
January 23, 1991	Letter from P. S. Tam to O. D. Kingsley (TVA), commenting on TVA's letter of November 7, 1990, which responded to NRC's request for additional information concerning Revision 4 of Q-list corrective action program (CAP) plan.
January 28, 1991	Summary of meeting of January 19, 1991, with TVA to discuss Watts Bar's stop-work order and proposed corrective actions.
January 29, 1991	Letter from P. S. Tam to O. D. Kingsley (TVA), requesting response to individual plant evaluation by November 30, 1991, to support closure of design alternatives regarding severe accident mitigation.

February 5, 1991	Summary of meeting of January 31, 1991, with TVA to discuss licensing status.
February 6, 1991	Letter from P. S. Tam to O. D. Kingsley (TVA), forwarding interim safety evaluation documenting need for additional information regarding TVA's response to NUREG-1232.
February 6, 1991	Summary of meeting of January 29, 1991, with TVA to discuss seismic classification.
February 11, 1991	Letter from P. S. Tam to O. D. Kingsley (TVA), forwarding safety evaluation accepting TVA's July 31, 1990, Revision 3 to CAP plan for facility replacement items program.
February 19, 1991	Letter from P. S. Tam to O. D. Kingsley (TVA), informing TVA of April 8, 1991, site audit to address design calculations for Seismic Analysis CAP implementation and civil structure elements evaluation program.
February 19, 1991	Notice of meeting of March 13, 1991, with TVA to address licensing status.
February 22, 1991	Letter from B. A. Wilson to O. D. Kingsley (TVA), forwarding summary of February 7, 1991, meeting with TVA regarding Unit 2 security restart items and permanent security upgrade for Browns Ferry, Sequoyah, and Watts Bar.
March 5, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), advising TVA that revision to welding CAP and proposed FSAR page changes are acceptable.
March 8, 1991	Letter from B. A. Wilson to O. D. Kingsley (TVA), confirming March 12, 1991, management meeting in Atlanta, Georgia, to discuss status of actions to address issues of work control and work quality.
March 12, 1991	Notice of meeting of April 10, 1991, with TVA to discuss implementation status of several Civil Engineering CAPs.
March 20, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), confirming site review planned for April 16, 1991, to address issues concerning Welding CAP.
March 20, 1991	Summary of meeting of March 13, 1991, with TVA to discuss licensing status.
March 20, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), informing TVA of development of comments for proposed additional systematic records review per March 11, 1991, telephone conversation.
March 21, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), requesting additional information regarding safety and relief valve discharge piping analysis.

March 22, 1991	Letter from S. A. Varga to D. A. Nauman (TVA), forwarding integrated design inspection (IDI) report 50-390/91-201 (January 7-18, 1991, and February 4-8, 1991).
March 25, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), approving of pages in Amendment No. 63, FSAR Chapter 15, regarding revised Westinghouse methodology on dropped rods.
March 27, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), confirming planned visit to facility to perform walkdown of pressurizer surge line, per Bulletin 88-11, "Pressurizer Surge Lines Thermal Stratification."
March 27, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), informing TVA that its April 2 and July 30, 1990, letters for Prestart Testing CAP are acceptable.
March 28, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), forwarding safety evaluation criteria used by TVA for evaluating refueling water storage tank.
March 28, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), forwarding safety evaluation regarding facility bypassed and inoperable status indication system. Design is in conformance with Regulatory Guide 1.47.
March 29, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), accepting proposed page changes to FSAR Section 3.2.2.5, regarding Heat Code Traceability CAP submitted by July 31, 1990, letter.
March 29, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), forwarding safety evaluation regarding removal of upper head injection system per September 17, 1986, letter.
April 3, 1991	Letter from B. A. Wilson to D. A. Nauman (TVA), forwarding summary of March 12, 1991, management meeting with TVA regarding corrective actions for work control and quality issues.
April 9, 1991	Notice of meeting of April 24, 1991, with TVA to discuss status of three Civil Engineering CAPs.
April 15, 1991	Letter from F. J. Hebdon to D. A. Nauman (TVA), forwarding safety evaluation on TVA's CAP deviation process.
April 17, 1991	Summary of meeting of April 10, 1991, with TVA to discuss licensing status.
April 23, 1991	Letter from F. J. Hebdon to D. A. Nauman (TVA), forwarding SER Supplement No. 6 (NUREG-0847) concerning operation of facility.

April 25, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), forwarding safety evaluation concluding that CAP plan to resolve various cable issues is acceptable. Information will be incorporated in SSER 7.
May 7, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), forwarding request for additional information on SSER 6 Outstanding Issue 19(i) regarding dynamic and static load on main steam safety valves.
May 15, 1991	Notice of meeting of May 22, 1991, with TVA to discuss licensing status.
May 17, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), forwarding safety evaluation granting TVA's request of September 26, 1985, for deviation for Section III.L.2.d of Appendix R (10 CFR Part 50) regarding use of steam generator saturation temperatures in auxiliary control room.
May 21, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), forwarding supplemental safety evaluation accepting containment cooling system and hot standby as acceptable safe-shutdown mode following a main steamline break.
May 24, 1991	Letter from P. S. Tam to D. A. Nauman (TVA), informing TVA of site visit June 12-14, 1991, regarding review of physical security plan.
May 30, 1991	Notice of meeting of June 27, 1991, with TVA to discuss licensing status.
June 4, 1991	Summary of meeting of May 22, 1991, with TVA to discuss licensing status.

TVA Letters

June 21, 1985	Letter from J. Domer to E. Adensam (NRC) concerning certification of the Watts Bar Technical Specifications.
September 19, 1985	Letter from J. Domer to NRC regarding removal of upper head injection system at Unit 2.
January 10, 1986	Letter from J. Domer to B. Youngblood (NRC) regarding the uncontrolled rod withdrawal event.
May 14, 1986	Letter from R. Gridley to B. Youngblood (NRC) concerning the uncontrolled rod withdrawal event.
September 17, 1986	Letter from R. Gridley to NRC concerning elmination of upper head injection system.
July 26, 1988	Letter from R. Gridley to NRC providing marked-up pages on upflow conversion and upper head injection.

August 17, 1989	Letter from M. Ray to NRC regarding main steamline breaks in ice-condenser plants.
November 3, 1989	Letter from M. Ray to NRC regarding main steamline breaks in ice-condenser plants.
October 19, 1990	Letter from E. G. Wallace to NRC forwarding "Watts Bar Nuclear Plant (WBN) Units 1 and 2--Seismic Design for Certain Safety-Related Steel Tanks."
December 20, 1990	Letter from E. G. Wallace to NRC revising June 29, 1989, response to NRC Bulletin 88-004, "Potential Safety-Related Pump Loss."
December 21, 1990	Letter from E. G. Wallace to NRC responding to Generic Letter 90-06 regarding resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," and Generic Issue 94, "Additional Low-Temperature Overpressure Protection for LWRs," per 10 CFR 50.54(f).
December 21, 1990	Letter from E. G. Wallace to NRC forwarding "Review of Integration of Engineering Assurance Functions Into Nuclear QA and Nuclear Engineering Part 3," and "Nuclear QA--Nuclear Engineering Action Plan to Address Nuclear Manager's Review Group Report R-90-04 NPS."
January 3, 1991	Letter from E. G. Wallace to NRC informing staff that the capability for the continuous collection of gaseous effluents, including implementing procedures and system upgrades, will be in place before fuel load, per SSER 5.
January 3, 1991	Letter from E. G. Wallace to NRC responding to request for additional information dated October 19, 1990, regarding various updates to FSAR Chapter 12.
January 18, 1991	Letter from E. G. Wallace to NRC forwarding Revision 1 to "TVA Nuclear QA Plan."
January 28, 1991	Letter from M. O. Medford to NRC documenting information presented at December 12, 1990, meeting, providing additional information on topics that arose during discussions on QA records, and enclosing responses to specific NRC comments and recommendations.
January 29, 1991	Letter from E. G. Wallace to NRC responding to NRC question on basis for design-basis accident spectra for steel containment vessel, per NRC audit of Amendment No. 64 to FSAR.
January 31, 1991	Letter from E. G. Wallace to NRC revising TVA's December 10, 1984, and August 22, 1985, responses to IE Bulletin 79-02 regarding pipe support base plate design using concrete expansion anchors, per NRC's request of June 28, 1985.

February 26, 1991	Letter from E. G. Wallace to NRC forwarding "Microbiologically Induced Corrosion Program Report." Report describes program for detection, assessment, and control of microbiologically, induced corrosion (MIC) at plant. List of commitments and summary of January 10, 1991, meeting on MIC also enclosed.
February 28, 1991	Letter from E. G. Wallace to NRC forwarding fitness-for-duty program performance data for July-December 1990. Performance data and summary of management actions for plant sites also enclosed.
March 7, 1991	Letter from E. G. Wallace to NRC forwarding "Watts Bar Training Department Initial Simulator Certification." Training began on simulator in May 1988 and initial certification testing was completed in January 1991.
March 15, 1991	Letter from E. G. Wallace to NRC notifying staff of rescheduling of 1991 emergency exercise for Sequoyah and Watts Bar plants.
March 26, 1991	Letter from M. O. Medford to NRC notifying staff that previously docketed implementation schedule for various plant commitments and submittals will require reassessment as a result of construction stop-work order.
April 1, 1991	Letter from E. G. Wallace to NRC forwarding proposed change to Revision 1 to QA Program Plan TVA-NQA-PLN89-A, revising Subsection 4.1.5 regarding new generation and Bellefonte construction. Change will be submitted in TVA's annual plan update.
April 5, 1991	Letter from E. G. Wallace to NRC informing staff of completion of all but one commitment regarding corporate nuclear performance plan, per Appendix 8 and May 5, 1989, submittal of Revision 6 to that plan.
April 9, 1991	Letter from E. G. Wallace to NRC forwarding a list of commitments regarding TVA's test program for evaluating shallow undercut anchors for shear. Test scheduled for completion by July 15, 1991.
April 11, 1991	Letter from E. G. Wallace to NRC requesting authorization to use alternative to Section III, Subsection ND-2000 of ASME Boiler and Pressure Vessel Code to radiograph longitudinal seams in fittings supplied by Tube Line Corporation.
April 16, 1991	Letter from E. G. Wallace to NRC submitting copy of affidavit certifying that copy of FSAR Amendment No. 65 to Watts Bar plan was served to individual.
April 18, 1991	Letter from E. G. Wallace to NRC forwarding "Watts Bar Nuclear Plant Pipe Support Minimum Design Load Evaluation Phase I and Phase II Report."

April 18, 1991	Letter from E. G. Wallace to NRC forwarding Revision 2 to Topical Report TVA-NPOD89-A, "Nuclear Power Organization Description."
April 18, 1991	Letter from E. G. Wallace to NRC clarifying position with respect to criteria to be used in performing reanalysis of Category I civil structures.
April 30, 1991	Letter from E. G. Wallace to NRC forwarding Volumes 1-5 of "Supplemental Information--TVA's Compliance to 10 CFR 50.49--Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."
May 8, 1991	Letter from E. G. Wallace to NRC forwarding TVA's response to requests for additional information to enable NRC to continue its review of FSAR Amendment Nos. 54-64. Open issues include: safety classification of cable trays and conduits, conduit damping, and equipment qualification.
May 10, 1991	Letter from E. G. Wallace to NRC supplementing TVA's October 13, 1988, letter regarding additional examples of welding activities performed per ASME Section XI or by non-stamp holder. Any future anomolies noted with N-5 data package will be resolved via applicable site corrective action procedure.
May 10, 1991	Letter from E. G. Wallace to NRC forwarding TVA's response to NRC letter of March 20, 1990, regarding CAP on QA records, to address NRC comments on TVA's additional systematic record review document.
May 16, 1991	Letter from M. O. Medford to NRC requesting that NRC grant construction permit extensions for Unit 1 to December 31, 1993 and Unit 2 to June 30, 1997. TVA has implemented a comprehensive plan consisting of CAPs for Unit 1. Construction activities were halted in order to improve work control.
May 16, 1991	Letter from E. G. Wallace to NRC clarifying position on Item 2 of NRC letter of January 9, 1991, regarding Generic Letter 90-06, "Resolution of Generic Issue 70, 'PORV and Block Valve Reliability,' and Generic Issue 94, 'Additional Low Temperature Overpressure Protection for LWRs'."
May 31, 1991	Letter from E. G. Wallace to NRC forwarding response to NRC interim supplemental SER regarding performance plan on master fuse list program.

APPENDIX B

BIBLIOGRAPHY

Argonne National Laboratory

March 31, 1989, Letter from William T. Sha (ANL) to Bernard L. Grenier (NRC), transmitting the report, "COMMIX Analysis of a Main Steam Line Break in the Catawba Lower Containment."

Institute of Electrical and Electronics Engineers

Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations" (ANSI N42.7-1972).

Standard 387-1977, "Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations."

U.S. Nuclear Regulatory Commission

August 22, 1983, Letter from Cecil O. Thomas (NRC) to E. P. Rahe, Jr. (Westinghouse), transmitting "Acceptance of Licensing Topical Report WCAP-8821 (P)/8859 (NP), 'TRANFLO Steam Generator Code Description,' and WCAP-8822 (P)/8860 (NP), 'Mass and Energy Release Following a Steam Line Rupture'."

April 1, 1991, Letter from Gary Holahan (NRC) to W. J. Johnson (Westinghouse), transmitting SER on "Acceptance for Referencing of Licensing Topical Reports WCAP-10986P/10987, and WCAP-10988P/10989."

Westinghouse Corporation

WCAP-7907, "LOFTRAN Code Description," April 1984.

WCAP-8170, "Calculational Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD)," June 1984.

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WCAP-10986P (Proprietary), WCAP-10987 (Non-Proprietary), "Ice Condenser Drain Test Results, Data Analysis, and Development of Drain-Flow Models for the LOTIC-III Ice Condenser Code," November 1985.

WCAP-10988 (Proprietary), WCAP-10989 (Non-Proprietary), "COBRA-NC, Analysis for a Main Steam Line Break in the Catawba Unit 1 Ice Condenser Containment," November 1985.

APPENDIX E

PRINCIPAL CONTRIBUTORS

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APPENDIX G

ERRATA TO WATTS BAR SAFETY EVALUATION REPORT, SUPPLEMENT 6

Section	Page	Change
Appendix E	1	The following names were inadvertently omitted from the NRC Technical Contractors listing: C. Costantino, City College of New York T. Tsai, NCT Engineering, Inc. A. Unsal, Harstead Engineering Associates

APPENDIX P

SAFETY EVALUATION: CORRECTIVE ACTION PROGRAM FOR CABLE ISSUES*

*Previously issued as enclosure to letter from P.S. Tam (NRC) to D. A. Nauman (TVA), April 25, 1991.

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

ENCLOSURE

SAFETY EVALUATION BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

CORRECTIVE ACTION PROGRAM PLAN FOR CABLE ISSUES

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-390

1.0 INTRODUCTION

The employee concerns program for the Watts Bar Nuclear Plant (WBN) identified areas where inadequate practices of cable installation may have caused damage to installed cables. TVA conducted a review of the cable installation practices at the Sequoyah Nuclear Plant (SQN) and the Brown Ferry Nuclear Plant (BFN). Based on this review, TVA agreed that cable installation practices may have caused damage to cables and presented its evaluation program to ensure cable integrity at these plants. The staff reviewed the TVA program and agreed with the resolution to the cable installation issues for SQN and BFN. The staff's evaluation for SQN and BFN is documented in NUREG-1232, Vol. 2 dated May 1988, and NUREG-1232, Vol. 3, Supplement 2, dated January 1991. TVA, by letter dated June 27, 1989, issued a Corrective Action Program (CAP) plan for cable issues, Revision 1, for WBN, which was based on conclusions reached at SQN and BFN in resolving similar issues. The staff reviewed the CAP and found it unacceptable. The staff review is documented in NUREG 1232, Volume 4 dated January, 1990. The principal reason for not accepting the CAP was the fact that pulling damage was found on cables which were removed from a conduit for inspection to resolve an employee concern (that heat from a welding arc near the subject conduit may have caused damage to cables). TVA sent these cables to University of Connecticut (U-CONN) to determine the root cause of the damage. The staff subsequently met with TVA on November 17, 1989, February 15 and 16, May 22, August 1 thru 3, September 20 and 21, and October 10 and 11, 1990, to discuss a revised program for resolving various cable installation issues. Based on those discussions, TVA submitted a revised program by letters dated December 12, 1989, June 15, July 31, October 11, and November 5, 1990. The revised program is detailed in the following evaluation section.

2.0 EVALUATION

2.1 Cable Issues In The CAP

TVA has identified the following technical areas related to cable issues in the Corrective Action Program (CAP):

2.1.1 Cable Pullby

In June of 1989, to resolve an employee concern, TVA removed the cables from a conduit run in the reactor protection system of Unit 2 to inspect for damage. This conduit was selected in response to an employee concern that a welding arc that struck the conduit during construction may have damaged cables in the conduit. When the cables were removed, significant damage was found in the insulation of some cables. However, this damage was not attributed to heat generated by the alleged welding arc. The damage was principally attributed to the pulling stresses exerted during the initial installation of the cables.

In order to fill a conduit, pull cords were used by Watts Bar personnel to pull additional cables through the conduit over the top of existing ones in the conduit (pullby). Potentially, this practice can cause damage to the existing cables from the sawing action generated by the pull cords and by the cables themselves as they are pulled over the existing cables. Usually, damage can be avoided by using adequate amounts of lubricants, by controlling pulling tension, by choosing appropriate pull cords, by controlling the distance between pull points, and by minimizing the number and angle of bends allowed in the conduit run. Currently, the industry standards provide no specific guidance for performing multiple pulls of cables in conduits. The concerns raised by TVA employees and the NRC staff have heightened industry interest in this subject.

To assess the adequacy of cable installation at its nuclear facilities, TVA instituted programs for corrective action. At Watts Bar, overlays of the damaged cables on plant isometric diagrams of conduit runs have indicated that cables appear to have been damaged at locations of the conduit runs where pull tensions and side wall bearing pressures (SWBPs) have exceeded certain safe threshold values (high risk). The TVA program for corrective action calls for replacement of cables that have exceeded the threshold values of SWBP. SWBP values are calculated as a function of the physical parameters of the cables and the conduit configuration. TVA's cable installation procedures (G-38) included conservative values of SWBP that the cable installation crews may not have followed at the time of major construction at Watts Bar. At a meeting on November 17, 1989 between the staff and TVA, TVA proposed a program for resolving the cable pullby issues, and by letter dated December 12, 1989, submitted the program for staff review. The submittal included the U-CONN report on the damaged cables which were discovered while cables were being removed to resolve the employee concern regarding possible damage due to the welding arc. U-CONN determined that the root cause of damage was cable pullby. TVA determined that cables that have not exceeded the safe threshold values (low risk) would not have experienced damage and are acceptable without any further action. The staff did not accept this determination, because the staff was concerned that the threshold value for damage may not have been conservatively defined. Therefore, during the meeting of February 15 and 16, 1990, the staff suggested that TVA either hi-pot (high-potential) test a sample (in the low risk population) of 20 worst-case conduits from the v1/v2 voltage level and 20 worst-case conduits from v3/v4 voltage level, or remove the cables for visual inspection to assure that cables are not damaged by cable pullby. During the meeting of May 22, 1990, TVA agreed to hi-pot test the 20 worst-case conduits of each group, and subsequently documented its intent in a submittal dated June 15, 1990.

TVA's program stipulated that cable damaged from pullby that failed the hi-pot testing of the worst-case sample would require replacement of all cables in that conduit and the sample would increase to twenty more worst-case below the original sample. TVA's program further stipulated, however, that cables in conduits that passed the test, in the sample, at higher ranking from those identified to have the damage would be accepted as undamaged cables. The staff disagreed with this stipulation and asked TVA to replace or pull back for inspection all cables ranked above the damaged cables that passed the test. The staff was concerned that since the test can only determine gross damage to cables, cables that passed the test at higher ranking may have unacceptable damage. This has become a moot point however, since TVA has completed testing of all worst case conduits and has identified no cable failure from pullby. However, recently the staff became aware that TVA has not included spare and abandoned cables in the test program. Since the pullby concern affects all cables in the conduit, the staff requested TVA to test all those cables unless it was determined that they were abandoned because of known damage to those cables. Hence this issue will remain open until all testing has been completed.

In its submittal of June 15, 1990, TVA further proposed to include some high-risk category conduits in the low-risk population. The staff disagreed with TVA's proposal because as indicated previously, hi-pot testing at voltage levels agreed upon can only detect gross damage to cables. During the meeting of August 1-3, 1990, TVA presented its argument to remove five conduits from the high risk population and include them in the test sample. TVA's argument was based on the fact that the calculated value of SWBP for these conduits was not very far from the low-risk value and also assumptions used in the calculation are very conservative. The staff disagreed with TVA for one conduit and TVA agreed to retain that conduit in the high-risk population.

The staff also expressed the concern that TVA was not using the recommendation of the IEEE-400 for hi-pot testing which requires the use of negative polarity DC. TVA agreed to use the negative polarity DC on future hi-pot testing and the staff did not require the repeat of the test on the one conduit which was conducted with positive polarity DC. Based on the above, the staff finds TVA's program to resolve the cable pullby issue acceptable, except for the issue related to spare and abandoned cables.

2.1.2 Cable Jamming

Cable jamming can occur when the ratio of the inside diameter of a conduit to the diameter of one cable in a three-cable pull has a value close to 3.0, causing the cables to jam or wedge in the conduit as the cables are pulled around a bend. Jamming is most likely to occur when cables are pulled around a bend rather than when being pulled in a straight run. The ratio of the diameter of the conduit to the diameter of the cable is called the jam ratio. The critical jam ratio (2.8 D/d 3.1) must be avoided in order to remove the concern for jamming. TVA did not take jam ratio into account during the sizing of the conduit and thus could not assure that cable damage has not occurred because of jamming. TVA originally bounded this issue by the results of the SQN investigation. However, when cable damage was identified at WBN, a separate investigation was planned to address this issue at WBN. The present

TVA approach for resolving the cable jamming issue is to identify cables with jam ratio violation and visually inspect those cables that are being removed for other related concerns and that also violate the jam ratio. Based on the visual inspections (assuming no damage from jamming is identified), cables in other conduits ranked lower according to SWBP calculations will be bounded by the inspected cables. TVA will analyze, inspect, rework or test to confirm that no damage has occurred on higher ranked cables. The staff considers TVA's approach acceptable. However, acceptability of the installation will be determined by the sample size inspected. The sample inspected must be sufficiently large to allow a statistical inference to be made about the integrity of the overall installation. If damage is found, TVA will perform a root cause analysis and inform the staff about the finding. The staff at that point may perform a visual inspection of the damaged cables.

2.1.3 Cable Support in Vertical Conduits and Trays

TVA procedure did not properly address the issue of cable support in long vertical runs. Vertical cables tend to creep downward and pull on the upper horizontal cable section, causing high stresses at the 90° bend and cutting off the insulation. The 90° condulets located at or near the top of vertical run represent a major potential for damage to cables at Watts Bar, especially in harsh thermal and wet environments. Since the standard 90° condulets have radii of 1/16 to 1/8 inch, tension in cables passing through the condulets causes the potential for severe damage from indentation and cutting of jacket and insulation. By letter dated June 15, 1990, TVA submitted a plan to resolve the issue of cable support in vertical conduits and trays. In this letter TVA committed to comply with the requirements of NEC Article 300-19 (1987), "Supporting Conductors And Vertical Raceways." However, the staff expressed a concern with the analysis used by TVA to determine the allowable length of vertical drop in conduit before cable restraint is provided. The staff's concern relates to the credit taken by TVA for power cables for the frictional resistance provided by the horizontal run immediately proceeding the vertical drop. During the meeting of August 1-3, 1990 and in a letter dated October 11, 1990, TVA agreed not to take credit for frictional resistance of horizontal run and to provide additional restraints upstream of the first access point for conduits that exceed the NEC-300-19 values.

2.1.4 Cable Proximity to Hot Pipes

Cables are designed for 40-year life assuming certain ambient temperatures. However, when hot pipes run close to cables, the higher ambient temperature will degrade the cable insulation and shorten the life of the cables. NRC Information Notice 86-49 highlighted the potential for cable damage resulting from close proximity to hot pipes. In a letter dated June 15, 1990, TVA submitted a plan to address this issue. During the meeting of August 1-3, 1990, the staff questioned TVA's decision to exclude: (a) the effects of insulated pipes operating below 250°F, (b) 2-inch and smaller diameter pipes and (c) the uninsulated pipes operating up to 135°F. In its letters of October 11, and November 5, 1990, TVA has stated that it will document the basis of acceptability of cables based on walkdown evaluations, calculations and modifications. This analysis will take into account the lower operating temperature based on Sequoyah data instead of the design ambient conditions in various plant areas.

The staff feels that this approach will be acceptable if sufficient data exist to justify the methodology. The data should consist of sufficient number of temperature measurements taken at different times so that they could be used to obtain an average temperature profile. Also, the applicability of the data to WBN should also be justified. The staff will audit the data during a future inspection. The staff, at a later date when the plant is operational, may check the values used by TVA in this analysis.

The staff also expressed a concern regarding the relationship between the hot pipes analysis and cable ampacity analysis. Cable ampacity analysis determines the load carrying capability (amperes) of the cable based on cable size and various derating factors, e.g. flammastic, tray cover, fill, etc. Also, since a hot pipe analysis could affect the cable design rating and assigned load, it is important that the two analyses are compared to determine their effect on cable rating. During the above meeting TVA assured the staff that the analysis of the effect of the hot pipes on the cables takes into account the derating of cables based on the cable ampacity analysis.

2.1.5 Silicone Rubber Insulated Cable

The initial test program proposed by TVA for SQN, dated April 30, 1987, included hi-pot testing of silicone rubber insulated cables manufactured by American Insulated Wire (AIW), Rockbestos and Anaconda. Subsequent testing of cable samples revealed a significant number of failures in the AIW cables. TVA decided to replace all AIW cables. For the Rockbestos and Anoconda cables that remained in the plant, the staff agreed with TVA to an Equipment Qualification (EQ) testing to demonstrate 40-year qualified life. Sample cables manufactured by Anaconda and Rockbestos were removed from WBN and tested at Wyle Laboratories to SQN environmental conditions. The staff reviewed the test results and found these cables acceptable for SQN (letter dated July 21, 1989). TVA performed radiation dose calculation and demonstrated that the environmental conditions at WBN are enveloped by the environmental conditions at SQN. Therefore, the staff has concluded that based on the results of the tests conducted for SQN, TVA has adequately resolved the silicone rubber-insulated cable issue for WBN.

2.1.6 Cable Bend Radius

Based on various employee concerns and non-conformance reports, TVA determined that the minimum cable bend radius recommended by Insulated Cable Engineers Association (ICEA) has been violated at WBN. Excessive bending has the potential of damaging cables and adversely affects their performance. Damage can be caused by: (a) elongation stress to the insulation system which may reduce the qualified life of the cable, (b) interfacial disruptions of medium voltage cable's stress control layers of insulation and insulation shield, which may have likelihood of corona degradations, and (c) conductor creeping which will put radial stress on the insulation system. Items (a) and (c) apply to low-voltage cables, while all three items apply to medium-voltage cables.

By letter dated June 15, 1990, TVA submitted a program plan to resolve the issue of cable bend radius. Medium-voltage cables have been tested to establish the lower bound for bend radius. The lower bound is defined as the lowest value of cable bend radius that will be acceptable for rebending to higher value. All medium-voltage and low-voltage cables in harsh environment that do not meet the lower bound will be replaced, and all Class 1E medium-voltage cables in non-harsh environment that do not meet the lower bound will also be replaced. All cables which meet the lower bound bend radius and are inside containment and main steam valve vault and are required to be environmentally qualified will be retrained to the ICEA requirement. In other areas, cables will be retrained or used "as is" with less qualified life and will be accepted based on margin analysis and on a long-term program. The staff questioned TVA's intent to: (a) exclude multiconductor low-voltage cables from testing to establish lower bound of cable bend radius, (b) account for aging effects, and (c) not seek input from cable manufacturers regarding the acceptability of the test program used to establish lower bounds. In its letter of October 11, 1990, TVA has agreed to include these items in the program. For aging effects, TVA agreed to perform corona and load cycle tests on medium voltage cables. Therefore, the staff finds the TVA program for cable bend radius acceptable.

2.1.7 Cable Splices

A cable splice is used to join two or more field cables together or to join a field cable to equipment pigtails. At WBN, Raychem heat shrink tubing, Raychem kits and a limited number of Scotch 3M tapes are used for cable splices. However, based on NRC Information Notice 86-53 and SQN experience, the installed cable splices may not conform with the qualified configuration and materials tested by the vendors. By letter dated June 15, 1990, TVA submitted its program plan to resolve the issue of cable splices. In accordance with this program, TVA will replace all cable splices required for 10 CFR 50.49 circuits located in harsh environment that are not vendor-supplied. Vendor-supplied equipment with splices have been qualified by the vendor to the requirements of 10 CFR 50.49. TVA will replace all intermediate splices, in mild environment, that are susceptible to moisture intrusion from flood, pipe break or sprinkler system activation. Other ongoing activities, such as cable replacement due to pullby concerns, ampacity, or bend radius violations etc., will serve to determine whether TVA's records accurately identify all cable splices at the site. If the cable replacement activities identify a significant number (based on 95/95 confidence level) of undocumented splices, TVA will re-evaluate its program to assure that all cable splices are adequate. Based on the above, the staff finds the TVA program to resolve the cable splice issue acceptable.

2.1.8 Sidewall Bearing Pressure

Sidewall bearing pressure (SWBP) is the radial force exerted on the cable insulation at a bend while the cable is being pulled in a raceway or around a sheave. At WBN, SWBP was not properly addressed in the design and installation process and may have exceeded the allowable values. By letter dated June 15, 1990, TVA submitted its program plan to resolve the issue. TVA has performed testing to confirm that higher SWBP values would not affect the integrity of cables. The staff has previously reviewed the test report and requested additional information on the test program. In its letter of October 11, 1990, TVA committed to provide a response to the staff's request. TVA walked down 81 worst-case conduits and calculated the SWBP for these conduits. Only one conduit exceeded the new design limits established by the test results and TVA committed to replace the cables in that conduit. The staff asked TVA to walkdown an additional 40 conduits from the harsh environment to confirm that no other violations of SWBP are present. By letter dated November 5, 1990, TVA documented that the additional 40 conduits have been walked down and no violations of SWBP were observed. Therefore, the staff agrees with TVA's resolution of the issue.

2.1.9 Pulling Cable Through 90 Degree Condulet and Flexible Conduit

Concern of potential damage to cables in 90° condulets was raised, because of the small supporting surface the inside corners of condulets provide for cables under tension. The sharp inside corners can in time cut into the insulation, or the conductors can creep through the insulation, reducing the insulation level of the cable. TVA plans to evaluate the effects of the 90-degree condulets on silicone rubber insulated cables which are more susceptible to damage than cables with other types of insulation. Also, a selection criterion for the worst-case silicone rubber insulated cables requires that cables as a minimum should have two 90-degree condulets within their route. The staff agrees with the TVA program to resolve this issue.

Also, concerns were raised regarding flexible conduits used at WBN in the middle of a conduit run. Since the inside surface of a flexible conduit has overlapping corrugations, the entire surface of the cable pulled through a flexible conduit segment in a bend will be subjected to very high frictional forces that can severely tear the cable jacket and insulation. At the meeting of August 1-3, 1990 the staff requested TVA to provide a program for resolving the concern involving pulling cable through midroute flexible conduits. TVA plans to evaluate cables pulled through midroute flexible conduits which have been tested for pullby damage, and inspect cables removed because of other concerns to confirm that no damage was caused by the midroute flexible conduits. If a sufficient sample exists to make that determination, then this will resolve the issue. If a sufficient sample does not exist, TVA will perform additional walkdowns to visually inspect cables at access points to confirm that no evidence of physical damage exists from pulling through flexible conduits. The staff agrees with TVA's program to resolve this issue.

2.1.10 Computerized Cable Routing System (CCRS)

At WBN, TVA is using the CCRS to document information regarding cable routing. The information includes cable routing in trays and conduits, cable type, cable weight, cable splices, circuit function, cable separation etc. Concerns regarding the adequacy of CCRS have been expressed and documented in various CAQRs, employee concerns and NRC inspection reports for SQN. By letter dated June 27, 1989, TVA submitted a program plan to resolve these concerns. TVA plans to: (a) qualify the computer software, (b) verify existing data, (c) revise procedures for controlling data entry, revision, and utilization, (d) expand the data base to support other activities, and (e) validate the system. The staff agreed with the TVA approach but asked TVA to also validate the CCRS with cables being removed or inspected because of other issues. TVA has agreed to evaluate the cable routing of cables removed to further validate the CCRS. Therefore, the staff finds TVA's approach to resolution acceptable.

2.2 OTHER CABLE ISSUES

In addition to the issues identified in the cable issues CAP, other issues related to cables are being addressed by TVA. These issues are discussed below.

2.2.1 Cable Ampacity

The Institute of Nuclear Power Operations (INPO) findings on the Bellefonte Nuclear plant identified the deficiencies in the TVA design standards used to determine cable ampacity. These findings resulted in a discrepancy report which was applicable to all TVA Nuclear plants. TVA has modified its design standards based on various industry standards and test reports. By letter dated June 15, 1990, TVA submitted its program to assess cable ampacity at WBN. The TVA program is similar to that of SQN, which was previously accepted by the staff. However, during the meeting of August 1-3, 1990, the staff asked TVA to justify the basis for a derating factor of 25 percent for cable tray covers of greater than or equal to 10 feet with no derating for covers of less than 10 feet. The National Electric Code requires 5 percent derating for tray covers of more than six feet. By letters dated October 11 and November 5, 1990, TVA has agreed to either: (a) perform laboratory testing to support a derating factor of 25% for tray covers of 10 feet or, (b) determine new derating factors based on testing and apply the new factors to those trays with covers in excess of six feet and less than 10 feet. Cables that cannot meet their ampacity requirements based on the derating factors identified above will be further reviewed to determine corrective action, or (c) the derating factor of 25 percent would be applied for any tray cover in excess of six feet and determine the acceptability of cables. The staff agrees with the TVA approach to resolve the staff's concern on cable ampacity.

2.2.2 Pulling of Large Low-Voltage Cables Using Standard Condulets As Pull Points

Cable damage was discovered during an inspection performed at BFN. The damage consisted of a cut to the cable jacket and insulation on one conductor in a 3-conductor 400-MCM cable and was located inside of two back-to-back 3-inch standard LB condulets. The root cause of damage was determined to be the use of standard condulet bodies as pull points for multiple large single conductor cables. The inflexibility of these cables coupled with the high fill resulted in excessive congestion at the fitting which caused damage to the cables.

By letter dated June 15, 1990, TVA submitted its program to resolve the concern regarding pulling of large low-voltage cables using standard condulets as pull points. This issue was identified during the cable walkdown at BFN. The staff had previously reviewed the resolution of this issue at BFN and found it acceptable (NUREG-1232, Vol. 3, Supplement 2, dated January 1991). The resolution plan at WBN is similar to the one used at BFN and hence, the staff agrees with the TVA program to resolve this issue.

3.0 CONCLUSION

Based on review of TVA submittals, the staff finds the CAP plan for the cable issues acceptable, except for the issue of spare and abandoned cables (see Page 3). The staff will perform inspections to assure adequate implementation of the program. The staff will further supplement this safety evaluation when the inspections are completed and all the open items identified in these inspections are resolved.

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