TVA

Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381

JUL 14 1994

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	50-391

WATTS BAR NUCLEAR PLANT (WBN) - FINAL SAFETY ANALYSIS REPORT CHAPTER 14 - PROPOSED CHANGES

The purpose of this letter is to provide the NRC with copies of draft proposed changes (enclosed) for several test abstracts in FSAR Chapter 14 "Initial Test Program." These changes were discussed with NRR and Region II staff during a meeting at the WBN site on Wednesday, July 6, 1994, and are provided in order to facilitate the NRC's understanding of WBN's testing plans. TVA expects to address these changes by FSAR Amendment 88 currently scheduled to be submitted on or about July 27, 1994.

If you should have any questions, contact B. S. Schofield at (615)-365-1857.

Sincerely,

Dwight E. Nunn
Vice President
New Plant Completion
Watts Bar Nuclear Plant

Enclosure
cc: See page 2

210005

D030
1/1

JUL 14 1994

cc (Enclosure):

NRC Resident Inspector
Watts Bar Nuclear Plant
Rt. 2, Box 700
Spring City, Tennessee 37381

Mr. P. S. Tam, Senior Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852

U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

ENCLOSURE

FSAR Chapter 14 Change Packages Issued Since Amendment 86

Change Pkg No.	Summary
1068	Preop Phase - Clarify that Post Accident Sampling Facility (PASF) testing does not include analysis/time measurements
1073	Preop Phase - Clarification for loss of control air testing
1079	Preop Phase - Deletion of abstracts where alternate testing will be performed instead of PTIs; also includes clarification of some abstracts
1080	Preop Phase - Similar to 1079
1089	Power Ascension Phase - Clarification to 8 test abstracts
1089 S1-3	Power Ascension Phase - Supplements 1089
1096	Preop Phase - Clarifies 2 abstracts; deletes load testing of Common Station Service Transformers (CSSTs) based on transformer capacity
1097	Preop Phase - Clarifications for Containment Integrated Leak Rate Test (CILRT) in Chapters 6 and 14.

Total Number of affected test abstracts:

Preop Phase	(Table 14.2-1)	- 19 of 61
Power Ascension Phase	(Table 14.2-2)	- 8 of 35

50-390 TVA WATTS BAR

FINAL SAFETY ANALYSIS REPORT CHAPTER
14 - PROPOSED CHANGES

REC'D W/LTR DTD 07/14/94....9407220216

APPENDIX C

Page 1 of 1

1068 S00 PKG

FSAR CHANGE REQUEST

FSAR CHANGE PACKAGE NO. 1068 ______ Supp. No.

R3

I. Attached are proposed changes to FSAR pages with justification for each change:

SECTION, PAGE NO. FIGURE OR TABLE	REASON FOR CHANGE
1) See Attached 2) (Table 14.2-1, Sh-11) 3) 4) 5) 6) 7) 8) 9) 10)	deleted testing will be accomplished ~~to the post fuel load testing program~~ IN ACCORDANCE WITH NCC'S ASSOCIATED WITH ITEM II.B.3 OF ATTACHED NUREG-0737 SUMMARY. See attached NCO920054454. THIS NCO requires implementation prior to fuel load and includes demonstration that PAS sampling per Chem Manual Ch 13 can be accomplished (incl. analysis) within Rg 1.97 time limits per discussion w/G. Vickery (Chemistry) 3/28/94.

(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
~~N/A~~ CHEMISTRY	See Below "D/V" next to	OPS (D. Voeller)	3/28/94
ENGINEERING	See Attached		
TECH SUPPORT	See Attached		
OPERATIONS D/V	D S King		3/28/94
LICENSING	C W Touchstone	C W Touchstone	3/28/94

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☐ or DO NOT ☒ affect the SER.

EVALUATED BY: [signature] Name SUT ENGR Title 3/28/94 Date

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

NUREG-0737, Item II.B.3 (TVA status pkg to NRC, 2-27-94, T03940227815)

V. PREPARED BY: [signature] NAME AND SIGNATURE 3/28/94 DATE

APPROVED BY: [signature] NAME AND SIGNATURE 3/28/94 DATE

VI. Approved ☐ Rejected ☐
Reason for Rejection: ______

Licensing Engineer: ______ Date: ______

R3

APPENDIX C

Page 1 of 1

1068 S00 PKG

FSAR CHANGE REQUEST

FSAR CHANGE PACKAGE NO. 1068 ______ Supp. No. R3

I. Attached are proposed changes to FSAR pages with justification for each change:

SECTION, PAGE NO. FIGURE OR TABLE	REASON FOR CHANGE
1) See Attached	deleted testing will be accomplished ~~in the post fuel load testing program~~ IN ACCORDANCE WITH NCOS ASSOCIATED WITH ITEM II.B.3 OF ATTACHED NUREG-0737 SUMMARY
2)	
3)	
4)	
5)	
6)	
7)	
8)	
9)	
10)	

(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
~~N A~~ MD 3/28/94			
ENGINEERING	FA KOONTZ JR RMJ	[signature]	3-28-94
TECH SUPPORT	Candy L McCormick	[signature]	3/28/94
OPERATIONS A/R	D S King	[signature]	3/28/94
LICENSING			

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☐ or DO NOT ☒ affect the SER.

EVALUATED BY: [signature] (Name) SUT ENGR (Title) 3/28/94 (Date)

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: [signature] NAME AND SIGNATURE 3/28/94 DATE

APPROVED BY: [signature] NAME AND SIGNATURE 3/28/94 DATE

VI. Approved ☐ Rejected ☐ R3
Reason for Rejection: ______

Licensing Engineer: ______ Date: ______

1068 S00 PKG

FSAR CHANGE REQUEST

FSAR CHANGE PACKAGE NO. 1068 ______
Supp. No.

R3

I. Attached are proposed changes to FSAR pages with justification for each change:

SECTION, PAGE NO, FIGURE OR TABLE	REASON FOR CHANGE D/V
1) See Attached	deleted testing will be accomplished ~~[illegible]~~
2) (Table 14.2-1, 54-11)	~~[illegible]~~ in accordance with ~~[illegible]~~
3)	~~[illegible]~~
4)	summary. See attached
5)	NCO 920054454. This NCO requires
6)	implementation prior to fuel load
7)	and includes demonstration that
8)	PAS sampling per Chem Manual Ch. 13
9)	can be accomplished (incl. analysis) within the 1.97 time limits.
10)	per discussion w/ G. Vickery (Chemistry) 3/28/94.

(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
~~[illegible]~~ Chemistry	See Below "D/V" next to OPS		
Engineering	See Attached		
Tech Support	See Attached		
Operations D/V	D S King	[signature]	3/28/94
Licensing	C W Touchstone	C W Touchstone	3/28/94

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☐ or DO NOT ☒ affect the SER.

EVALUATED BY: [signature] Name SUT Engr Title 3/28/94 Date

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: [signature] NAME AND SIGNATURE 3/28/94 DATE

APPROVED BY: [signature] NAME AND SIGNATURE 3/28/94 DATE

VI. Approved ☐ Rejected ☐
Reason for Rejection: ______

Licensing Engineer: ______ Date: ______

R3

1 5 5 9 4 0 3 2 4 1 2 9

MAR 24 1994

1068 S00 PKG

Those listed,

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - NUREG-0737, POST TMI REQUIREMENTS - COMPLIANCE EVALUATION/CERTIFICATION OF COMPLETION REVIEW

This memorandum is requesting that each of the NUREG-0737 issues be reevaluated for compliance with the TVA/NRC docketed positions.

Many of the NUREG-0737 issues have been closed by the NRC since the 1985 timeframe. Before the NRC issues an operating license to WBN, TVA must certify that the design basis of the plant (including docketed correspondence to and from the NRC) remains accurate. Because of the significance of the NUREG-0737 items and the modifications to the plant design and procedures, each of the items that has not been closed by the NRC or has not had a closure package issued to the NRC since January 1, 1993 or has not had an open item status package assembled since January 1, 1994, must be reevaluated for commitment compliance and to determine if any remaining work associated with the issue exists.

Attachment 1 is a summary listing of each NUREG-0737 item. This list includes the NRC NRR/Region II status, any open commitment tracking numbers, current status and the responsible TVA manager. Attachment 2 provides a Compliance Evaluation/Certification of Completion Form for each of your assigned NUREG issues. Copies of the TVA/NRC docketed correspondence and the NUREG requirement for that issue are also attached.

Please evaluate each of your assigned issues for completion and compliance with docketed correspondence, remaining work to bring the plant in compliance with the requirements or commitments, and to determine if TVA's position has been revised and not yet submitted to the NRC for approval. Please complete the Compliance Evaluation/Certification of Completion Form, and the Form attachment, if needed, and acknowledge by your signatures that the issue has been evaluated, the current status is correct as provided, and that all activities to complete remaining work for that issue are identified and will be completed within the NRC approved scheduled milestone. All activities associated with the NUREG-0737 items must be completed before fuel load with the exception of certifying operability of the SPDS (Item I.D.2), which must be <u>functional</u> before fuel load.

Please complete the Compliance Evaluation/Certification of Completion Form and return to Site Licensing by April 15, 1994. Should you have any questions concerning this subject, please contact Becky Mays at extension 3855 in Site Licensing.

Bill

W. J. Museler
Site Vice President
FSB 1A-WBN

(see page 2 for Those listed)

TABLE 14.2-1

(Sheet 11 of 90) 1068 S00 PKG

POST ACCIDENT SAMPLING SYSTEM
TEST SUMMARY

OBJECTIVE

To demonstrate that samples required under post accident conditions can be obtained, properly handled and analyzed in a safe ~~and timely~~ manner.

PREREQUISITES

1. Supporting systems are operational as required.

TEST METHOD

1. Demonstrate the ability to obtain samples and analyze from the reactor coolant system, containment sump and containment atmosphere and transport of the samples to a transfer station or onsite laboratory.

2. Demonstrate that waste liquids go back to containment or to the radwaste system and that gas sample can be disposed of properly.

3. Verify liquid sample panel and containment air sample panel (CASP) carts/casks are operational.

ACCEPTANCE CRITERIA

Samples can be obtained from all sample locations at the required flowrates ~~and safely transported for onsite analysis or to a transfer point for offsite analysis and analyzed within the required timespan as described by FSAR Section 9.3.2.6.~~

ITEM	TITLE	NRR	REGION II	RO MANAGER(S)	TRACKING/ COMMITMENT NUMBER	COMPLETION DATE/STATUS
II.B.3	Postaccident Sampling	Open SSER 5, LC-19	Open Status 390/91-04	NE/Koontz (Benninghoff) Voeller	NCO 850404 093 NCO 850404 381 NCO 820253 023 NCO 920054 454	03/94 (Letter to NRC) Status Package to NRC 2/27/94
II.B.4	Training For Mitigating Core Damage	Resolved SER	Closed 390/85-08	McNair		
II.D.1	Relief & Safety Valve Test Requirements	ReOpened Letter 10/10/91	Closed 390/85-08	NE/Koontz (Wiggall)	NCO 850385 007	4/94 (Letter)
II.D.3	Relief & Safety Valve Position Indication	Resolved SER	Closed 390/84-35	NE/Brickey (Faulkner)		
II.E.1.1	Auxiliary Feedwater Evaluation	Resolved SER	Open Status 390/91-04	NE/Koontz (Wiggall)		Endurance Test during HFT. Status pkg to NRC 1/31/94
II.E.1.2	AFW Initiation & Flow Indication	Resolved SER	Closed 390/84-20	NE/Brickey (Faulkner)		
II.E.3.1	Emergency Power for Pressurizer Heaters	Resolved SER	Closed 390/84-20	NE/Brickey (Ackley)		
II.E.4.1	Dedicated Hydrogen Penetrations	Resolved SER	Closed 390/83-27	NE/Koontz (Benninghoff)		
II.E.4.2	Containment Isolation Dependability	Resolved SSER 5	Closed 390/84-35	NE/Koontz (Benninghoff)		
II.F.1.1	Accident Monitoring Instrumentation - Procedures	NO NRR REVIEW	Open Status 390/91-04	Koehl (Woods)	NCO 820253 035 NCO 850192 002	Procedures will be included in other parts of II.F.1
II.F.1.2a	Noble Gas Monitoring	Resolved SSER 5	Reopened 390/91-04	NE/Koontz (Metcalf) Woods Voeller	NCO 850192 002 NCO 850192 003 NCO 850192 005	System 90
II.F.1.2b	Iodine Particulate Sampling	Resolved SSER 6	Reopened 390/91-04	NE/Koontz (Metcalf) Voeller		Status Package in Site Review (System 90)

H068 SOC PKG
T03 940227 815

NRC OPEN ITEM - STATUS FORM

NRC Open Item No. NUREG-0737, ITEM II.B.3   Contact: G. E. Vickery - CHEM
D. G. Fickey - NE
E. T. Haston - NE
NAME   ORG

LICENSING CONTACT: John Lovell - X2074

NRC OPEN ITEM -- NUREG-0737 Item II.B.3 (TMI 80-RD-13), Post Accident Sampling Capability.

NUREG-0737 Item II.B.3 requirements were originally provided to the nuclear power industry in a letter dated October 31, 1980 (Tab 1). Item II.B.3 requires that the capability exist for personnel to promptly obtain a sample of the reactor coolant system (RCS) and containment atmosphere under accident conditions without incurring an excessive radiation exposure. Included with this item are specific requirements that a Post Accident Sampling System (PASS) must satisfy.

RESPONSE

WBN initially responded to NUREG-0737, Item II.B.3, in NRC submittal letter dated September 14, 1981 (Tab 2). In a letter dated October 29, 1981, (Tab 3), WBN provided a revised response. The NRC's review of these responses, as written in the SER (Tab 4), resolved 5 of the 11 criteria. The following six criterion were unresolved.

Criterion (2) — Provide a procedure for relating radionuclide and ionic species to estimated core damage.

Criterion (5) — Verify that chloride analysis can be completed within four days following an accident.

Criterion (8) — Demonstrate the capability of analyzing the grab samples and verify that equipment provided for backup sampling shall be capable of providing, at least, one sample per day for seven days following onset of the accident and, at least, one sample per week until the accident condition no longer exists.

Criterion (9)(a) — Verify that the sensitivity of onsite liquid sample analysis capability is such as to permit measurement of nuclide concentration in the range from approximately 1 ~~Ci~~ µCi/g to 10 Ci/g.

PEW 2/11/94

Criterion (9)(b) — Verify that provisions are available to restrict background radiation levels such that the sample analyses will provide results with a range of accuracy within a factor of 2.

Criterion (10)(a) Describe the procedures for onsite radiological and chemical analyses and provide the accuracy, range, and sensitivity of these analyses in an accident chemistry and radiation environment.

Criterion (10)(b) Provide information on frequency/type of testing to ensure long term operability of the PASS and on operator training requirements for post accident sampling.

Criterion (11)(a) Verify that the residues of sample collection will be returned to containment or to a closed system.

Criterion (11)(b) Verify that the ventilation exhaust from the sample station will be filtered with charcoal absorbers and high-efficiency particulate air filters.

In addition to resolving the above, submit data supporting the applicability of each selected analytical chemistry procedure or online instrument, 4 months before exceeding 5% power operation (SER, p. 9-18).

WBN responded to these items and provided the final response, in submittal letters dated September 20, 1983 (Tab 5), December 19, 1983 (Tab 6), and July 13, 1984 (Tab 7). NRC's review of these submittals, as noted in Supplement 3 to the SER, (SSER 3) (Tab 8), determined that the Watts Bar PASS met all of the NUREG-0737 Item II.B.3 criteria and, therefore, was acceptable. This acceptable status was based on the review of an interim core damage assessment procedure (Criterion 2) understood to be replaced later by a final version. The following actions remained to be completed for the PASS as noted in the above TVA submittals, the SER and, it's supplements:

A. Complete installation of the PASS.

B. Implement onsite radiological and chemistry analysis procedures for the PASS.

C. Provide a procedure to the NRC for estimating the degree of reactor core damage from measured and predicted post accident radionuclide concentrations from failed fuels.

D. Develop and implement a functional testing and personnel training program for the PASS.

E. Submit data supporting the applicability of each selected analytical chemistry procedure or online instrument.

F. Revise Technical Specifications.

Completed Actions

A. The PASS was installed on unit 1 by:

Engineering Change Notices (ECN):		Closed Work Plans (WP):
2343	installed PASS per WB-DC-40-39	2830, 3385, 4197, 4198, 4502, 3829, 2598, 2953, 2999, 2029, 2446, 4176, and 3233 (Tab 9)
4901	modify discharge piping on PASF panels	4392 (Tab 10)
5050	replace FSV-43-315 and modify drains	4629 and 4750 (Tab 11)
5062	install fire protection	4669, 4688 and 4721 (Tab 12)

The following modifications affecting the PASS were completed:

Design Change Notices (DCN):		Closed Work Plans (WP):
M-16349	Gas Analyzer Addition,	D-16349-01, 02, 03, 04 and 06 (Tab 13)
M-15982	containment penetrations and isolation valves	D-15982-01 and D-15982-02 (Tab 14)
P-02939	isolation of HVAC equipment	D-02939-01 (Tab 15)

F. Technical Specification 5.7.2.6 (Tab 16) requires a PASS Program which is described in Plant Administrative Instruction (PAI)-15.04, "POST-ACCIDENT SAMPLING PROGRAM" (Tab 17).

Additionally, SER Supplement 5 (Tab 18) also addresses that WBN has met all of the eleven criteria for this NUREG item. It also noted that WBN would be expected to provide a final procedure of estimating the degree of core damage to satisfy Proposed License Condition 19.

The PASS is described by System Description N3-43B-4001 (Tab 19) and the FSAR section 9.3.2 (Tab 20), with design criteria for the post accident sampling facility (PASF) found in WB-DC-40-39 (Tab 21).

TVA provided to the NRC corrected Regulatory Guide 1.97 status for the PASS in a letter dated 10/29/91 (Tab 22).

Remaining Actions

The NRC further reviewed this NUREG-0737 item in Inspection Reports (IRs) 390/84-74 (Tab 23), 390/85-20 (Tab 24), and 390, 391/91-04 (Tab 25). IR 85-20 closed Inspector Followup Items (IFIs) 390/84-74-01, "Pressure Correction and Line Losses for Containment Atmosphere Sampling System," and, 390/84-74-02, "Verification of Sampling Point for Containment Atmosphere Post Accident Sampling System." In IR 91-04, the inspector stated that Items 1 through 4 were noted by WBN as being ready for NRC review. Although item 1 has been addressed, and is included here for continuity, items 2 through 5, as listed in IR 390/91-04, have remaining actions:

Remaining Actions (continued)

1. the need for additional shielding in the sample room to be evaluated; This condition has been evaluated and determined to be adequate per the Mission Dose Calculation (Tab 26).

2. the need for evaluations of temperature and pressure corrections and line losses in sample lines; (IFI 390/84-74-01 was closed in IR 390/85-20.) Note, it was written in IR 390/85-20 that, TI-16 had been revised to provide for temperature correction when actually, TI-66 had been. Regardless, TI-66 has been allowed to expire past it's 2-year review period due to it's planned replacement by Chemistry Manual Chapter 13 which will provide the necessary temperature corrections.

3. the need to verify the containment atmosphere sample is actually taken from the containment; (IFI 390/84-74-02 was closed in IR 390/85-20.) Test Scoping Document TVA-65 (Tab 27) has been amended to require the subject verification. (See also FSAR Table 14.2-1, Sheet 11) (Tab-20). NOTE: TVA-65 will be de-activated before the Systems Plant Acceptance Evaluation, Phase I (SPAE I) and, the subject verification added to the system description, N3-43B-4001 (Tab-19).

4. the need for NRC review of Technical Instruction TI-66; (Chemistry Manual Chapter 13, currently in draft, will supersede TI-66.)

5. the need for NRC inspection after reaching five percent power.

Although the WBN PASS has been installed, modifications, outstanding system testing, turnover, training of personnel, and procedure upgrades exist which are scheduled for completion prior to Unit 1 fuel load. Safety Evaluation Report (SER) License Condition 19 (TAC M77543) remains unresolved which, will require submission of WBN's final version of the core damage assessment procedure (presently in draft).

A. Although major installation of the PASS components is complete, modifications which affect system testing and turnover to the plant are ongoing: (attached is Master Tracking System (MTS) status as of 2/17/94 for associated workplans): This item is tracked by NC0820253023.

Design Change Notices (DCNs):		Closed Work Plans (WPs):
DCN M-16243	thermal qualification of radiation sampling and monitoring lines (approx 47 open WP)	(approx 65 closed WP) (Tab 28)

Remaining Actions (continued)

B. The onsite radiological and chemistry analysis procedure for the PASS TI-66, "Post Accident Sampling and Analysis Methods" (Tab 29), is currently planned to be superseded by Chemistry Manual Chapter 13. The existing System Operating Instruction (SOI)-43.02, "POST ACCIDENT SAMPLING SYSTEM" (presently past it's 2-year review) (Tab 30) is planned to be revised and re-activated to provide operating instructions for the PASS by chemistry and operations personnel. This item is also tracked by NC0820253023.

C. Develop a procedure for estimating the degree of reactor core damage and submit to the NRC. This action is tracked by NC0850404381. Engineering has provided a draft of the final version which is being reviewed for approval. Once approved on site, it will be sent to the Fuels Department for review and revision to applicable procedures and, submitted to the NRC.

D. Develop and implement a functional ~~testing and personnel training~~ program for the PASS. This action is tracked by NC0920054454. Training of personnel for the PASS is addressed in PAI-15.04 and TRN-30, "NUCLEAR POWER TRAINING PROCEDURE" (Tab-17).

E. Submit data supporting the applicability of each selected analytical chemistry procedure or online instrument. This action is tracked by NC0850404093.

This item is in a "hold" status for Unit 2 and is being tracked by IFI 391/85-48-09. The Unit 1 PASS is scheduled for completion prior to Unit 1 fuel load.

EW 2/23/94

[signature] 12/23/94
awS 2-22-94 RADCON/CHEMISTRY MGR / DATE

CSS 2/22/94
SJ 2-23-94
[signature] 12-23-94
MODIFICATIONS MGR / DATE

2-18-94
2/17/94
DGF 2/17/94
Revd for FHK 2/18/94
[signature]
J.E. TORRZA FOR RMJ / 2-18-94
NUCLEAR ENGINEERING MGR / DATE

[signature] 12/27/94
LICENSING MGR / DATE

Attachments:

Tab 1 - NRC letter dated October 31, 1980 conveying NUREG-0737, Item II.B.3 requirements
Tab 2 - TVA to NRC letter dated September 14, 1981 (A27810914022)
Tab 3 - TVA to NRC letter dated October 29, 1981 (A27811029002)
Tab 4 - SER section 1.9 and 9.3.2
Tab 5 - TVA to NRC letter dated September 20, 1983 (A27830920007)
Tab 6 - TVA to NRC letter dated December 19, 1983 (A27831219022)
Tab 7 - TVA to NRC letter dated July 13, 1984 (A27840713005)
Tab 8 - SSER 3 section 9.3.2
Tab 9 - ECN-2343 and completed work plans 2830, 3385, 4197, 4198, 4502, 3829, 2598, 2953, 2999, 2029, 2446, 4176, and 3233
Tab 10- ECN-4901 and completed work plan 4392
Tab 11- ECN-5050 and completed work plans 4629 and 4750
Tab 12- ECN-5062 and completed work plans 4669, 4688 and 4721
Tab 13- DCN M-16349 and completed work plans D-16349-01, 02, 03, 04, and 06
Tab 14- DCN M-15982 and completed work plans D-15982-01 and D-15982-02
Tab 15- DCN P-02939 and completed work plan D-02939-01
Tab 16- Technical Specification 5.7.2.6
Tab 17- PAI-15.04 and TRN-30
Tab 18- SSER 5 and SSER 12
Tab 19- PASS System Description N3-43B-4001
Tab 20- FSAR section 9.3.2 and Table 14.2-1
Tab 21- TVA Design Criteria WB-DC-40-39
Tab 22- Regulatory Guide 1.97 transmittal dated 10/29/91
Tab 23- Inspection Report 390/84-74
Tab 24- Inspection Report 390/85-20
Tab 25- Inspection Report 390/91-04
Tab 26- Mission Dose Calculation WBNTSR-085 (B18930320253)
Tab 27- Test Scoping Document TVA-65
Tab 28- DCN M-16243 and MTS report of closed work plans
Tab 29- TI-66
Tab 30- SOI-43.02

cc: Regulatory Licensing Files, FSB 2K-WBN
RIMS, QAC 1G-WBN

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MSLC001  PU8      TRACKING AND REPORTING OF OPEN ITEMS SYSTEM          03/28/94
                              FRONT END INQUIRY                         12:22:29

ITEM NUMBER: NC0920054454          TYPE: BQ  PLANT: W   REV:      PGMSTAT: A
DESCRIPTION: IN REGARD TO WBN SER SUPPLEMENT 3(9.3.2),  THE APPLICANT HAS STATE
D THAT THE ACCURACY, RANGE, AND SENSITIVITY OF THE WATTS BAR PASS INSTRUMENTS A
ND ANALYTICAL PROCEDURES ARE CONSISTENT WITH THE RECOMMENDATIONS OF RG 1.97, RE
VISION 3 AND THE CLARIFICATION OF NUREG-0737, ITEM II.B.3. THE APPLICANT PROPOS
ES THAT EQUIPMENT USED IN POSTACCIDENT SAMPLING AND ANALYSIS BE CALIBRATED OR T
ESTED ANNUALLY AND RETRAINING OF OPERATORS FOR POSTACCIDENT SAMPLING BE PERFORM
ED (CONT ON REMARKS 01)                       SIG: N  REP: N DATE OPENED:  031792
TRAC ORG/GRP/SUPV: WBL REG TRP   ENTRY DATE...: 031792  P2KEY: PNC0920054454
RESP ORG/GRP/SUPV: WBO CEM DJV   ORG PRTY DATE: 110193 P2DATE: 070290
INIT ORG/GRP/SUPV: WBO CEM DJV           UNIT: 1
NRC REF.: SER S3.9.3.2                  OTHER:
KEYWORDS:
                                                         TYPES: LM
COMMENTS:

  PWL:   PF                                  IDNT(A/C): A PRIORITY DATE:  053194
STATUS:  00 Open           STATUS DATE:         CLOSURE RIMS:
SELOPT:  5 ITEM NUMBER: NC0920054454        TYPE: BQ PLANT: W

CLEAR==> CANCEL  PF3==> MENU  ENTER==> PROCESS   PF9==> RETURN TO SEARCH LIST
Online to   WLIC (07)    ¤4B#                            ¤F4 DOS     ¤F10 EXIT
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MSLC00B PU8 TRACKING AND REPORTING OF OPEN ITEMS SYSTEM 03/28/94
REMARKS INQUIRY 12:23:51

ITEM NO: NC0920054454 TYPE....: BQ PLANT: W
SEQ NO.: 01 DOC DATE: 031792 DOC TYPE: 4 RIMS NO:
DESCRIPTION: AS NEEDED. EVERY 6 MONTHS, ONE-HALF OF THE CHEMISTRY TECHNICIANS WILL BOTH OPERATE THE PASS AND ACTUALLY TAKES SAMPLES OF THE FLUIDS IN PERTINE NT SYSTEMS. AT THE SAME TIME, IDENTICAL SAMPLES WILL BE TAKEN IN THE HOT SAMPLE ROOM. THIS WILL VERIFY T HAT THE PASS IS FUNCTIONING PROPERLY. BY USING THIS TIMETABLE, THE OPERATOR WILL BE RETRAINED ON A YEARLY BASIS, AND THE PASS WILL BE TESTED EVERY 6 MONTHS.

SEQ NO.: 02 DOC DATE: 011393 DOC TYPE: RIMS NO:
DESCRIPTION: ESTABLISHED TRIOI ITEM CEM-920512.

SELOPT: B ITEM NUMBER: TYPE: PLANT: SEQ:

CLEAR/PF3==> MENU PF6==> NER PF7==> BACK PF8==> FORWARD ENTER==> PROCESS
Online to WLIC (07) ¤4B# ¤F4 DOS ¤F10 EXIT

1073 S00 PKG

FSAR CHANGE REQUEST

FSAR CHANGE PACKAGE NO. 1073 ______ Supp. No.

R3

I. Attached are proposed changes to FSAR pages with justification for each change:

SECTION, PAGE NO. FIGURE OR TABLE	REASON FOR CHANGE
1) SEE ATTACHED	
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(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
ENGINEERING			
OPERATIONS		[signature]	4/4/94
TECH SUPPORT	Landa L. McCormick		6/23/94
LICENSING	Worthey	[signature]	

SBN 4-4-94 CNM 6/24

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☐ or DO NOT ☒ affect the SER.

EVALUATED BY: [signature] Name — SUP ENGR Title — 4/1/94 Date

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: [signature] NAME AND SIGNATURE — 4/1/94 DATE

APPROVED BY: [signature] NAME AND SIGNATURE — 4/1/94 DATE

VI. Approved ☐ Rejected ☐
Reason for Rejection:

Licensing Engineer: ______ Date: ______

R3

APPENDIX C

Page 1 of 1

1073 S00 PKG

FSAR CHANGE REQUEST

FSAR CHANGE PACKAGE NO. 1073 ______ Supp. No. R3

I. Attached are proposed changes to FSAR pages with justification for each change:

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(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
ENGINEERING	[signature]	D.A. KULISCK	4/5/94
OPERATIONS			
TECH SUPPORT			
LICENSING			

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☐ or DO NOT ☒ affect the SER.

EVALUATED BY: [signature] Name   SUT ENGR Title   4/1/94 Date

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: [signature] NAME AND SIGNATURE   4/1/94 DATE

APPROVED BY: [signature] NAME AND SIGNATURE   4/1/94 DATE

VI. Approved ☐ Rejected ☐
Reason for Rejection: ______

Licensing Engineer: ______ Date: ______

R3

APPENDIX C

Page 1 of 1

1073 S00 PKG

FSAR CHANGE REQUEST

FSAR CHANGE PACKAGE NO. 1073 ____ Supp. No. R3

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II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
ENGINEERING	FAKoontz Jr for KMJohnson	[signature]	4-13-94
OPERATIONS			
TECH SUPPORT			
LICENSING			

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☐ or DO NOT ☒ affect the SER.

EVALUATED BY: [signature] Name SUT ENGR Title 4/1/94 Date

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: [signature] NAME AND SIGNATURE 4/1/94 DATE

APPROVED BY: [signature] NAME AND SIGNATURE 4/1/94 DATE

VI. Approved ☐ Rejected ☐

Reason for Rejection: ____

Licensing Engineer: ____ Date: ____ R3

FSAR Chapter 14
Reason for Changes

Section 14.2.7.7

To clarify the exceptions being taken to RG 1.68.3

Table 14.2.1, Sheet 86 and 87

To properly reflect the test commitments as outlined in revised Section 14.2.7.7 above

7. RG 1.68.3, "Preoperational Testing of Instrument and Control Air Systems"

Discussion: Preoperational testing of the Instrument and Control Air Systems will comply with RG 1.68.3, April 1982, with the following exceptions:

a. Regulatory Position C.8

affected components on an individual valve basis

Auxiliary Control Air System ~~loads~~ will ~~not~~ be tested to verify their response to a sudden loss of system pressure. NRC concurrence with this exception is reflected in correspondence from R. C. Lewis to H. G. Parris dated February 28, 1984.

b. Regulatory Position C.8 see attached sheet for addit

Non-safety related loads will not be tested to verify their response to a loss of system pressure as part of the Auxiliary Control Air System preoperational test. The Auxiliary Control Air System does not supply air to non-safety related loads. However, non-safety related air operated components will be tested on a component basis to verify proper response to a loss of air condition. Delete

c. Regulatory Position C.11

Functional testing to demonstrate operability of compressed air system loads under increased pressure conditions will not be performed. The safety evaluation of the system indicates that the system is adequately designed to prevent system overpressure as described in Section 9.3.1.3. The maximum pressure rating of the most limiting component of the system piping, valves, and equipment (up to the end user pressure regulator) was determined to be at least 20 psi higher than the system safety valve setpoint plus 10% accumulation. This ensures that there is adequate protection against overpressurization.

8. RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," Revision 1, January 1977.

Discussion: The potential for chlorine to pose a hazard to main control room (MCR) operators due to onsite storage spills or transportation incidents in the vicinity of the site is analyzed in FSAR Section 6.4.4.2. The analysis concluded that no hazard to control room habitability is posed by chemicals stored on site, offsite within a 5-mile radius, or transported by the site by barge, rail, or road within a 5-mile radius. Therefore, the requirements of RG 1.95, Revision 1, (January 1977) "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release" do not apply.

b. Regulatory Position C.8

Control Air System affected components (safety related valves only) will be tested to verify their response to a loss of system pressure. The sudden loss of air pressure will be performed on an individual valve basis. Non-safety related air operated components will be tested on a component basis to verify proper response to a loss of air pressure.

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TABLE 14.2-1

COMPRESSED AIR SYSTEM
TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Compressed Air System to provide regulated air that is clean, dry and oil-free to instrumentation and control loads during normal plant operation and to vital equipment required for safe shutdown under design basis event conditions.

PREREQUISITES

1. AC and DC electrical power supplies are available.

2. The system has been blown-down and verified to be clean in accordance with approved cleanliness standards.

3. Cooling water is available.

4. The system filters are installed and dryers loaded with desiccant.

TEST METHOD

1. Simulate signals required to verify proper operation of automatic controls, interlocks, and alarms including:

   a. Automatic isolation of the Auxiliary Air System from the Station Air System on loss of air pressure in the Station Air System.

   b. Automatic start of the Auxiliary Air Compressors on receipt of a low auxiliary control air header pressure signal.

2. Verify automatic operation of air dryers for one regeneration cycle.

3. Measure the dewpoint of Station Control Air and Auxiliary Control Air.

4. Operate the station air compressors and auxiliary air compressors to verify proper operation of unit controls and cooling water. Verify proper compressor capacity and pressure.

see attached sheet for addition

5. Delete Perform a slow loss of Auxiliary Control Air System pressure by isolating as many branch lines as practical with the affected loads in their normal operating position to verify response of supplied loads.

6. With the Auxiliary Control Air System operating in a steady state condition, operate at least two of the highest demand loads supplied from the same train, simultaneously.

7. see attached sheet for addition

TEST METHOD

5. Perform a sudden and gradual loss of Auxiliary Control Air System pressure, with the affected components in their normal operating position, to verify that the affected components respond correctly.

7. Perform a sudden and gradual loss of Control Air System pressure, with the affected components in their normal operating position, to verify that the affected components respond correctly.

TABLE 14.2-1

COMPRESSED AIR SYSTEM
TEST SUMMARY

ACCEPTANCE CRITERIA

1. The station air compressors, auxiliary air compressors and associated air dryers properly operate to provide compressed air that meets or exceeds design requirements for air-flow and dewpoint and is within design pressure limits as described in FSAR Section 9.3.1.

2. Automatic controls including system isolation features, interlocks, and alarms operate in accordance with design drawings.

3. Auxiliary Control Air ~~loads~~ (affected components) respond properly to a ~~slow~~ loss of system air pressure as described by design documents.

4. Simultaneous operation of the two highest demand loads supplied from each train will not cause unacceptable system pressure transients as described in applicable design documents.

5. Control Air affected components (safety related valves only) respond properly to a loss of system air pressure as described by design documents.

FSAR PTI CHANGES ASSOCIATED WITH CR1079, 1080

* **PTI 40-01** - Station Drainage

    Perform as a combination of Component Level testing and/or ATI
    Delete FSAR Test Summary (Sheet 12)

* **PTI 39-03** - Fan Door Testing

    Perform as a combination of ATI and/or Component Level testing
    Modify FSAR Test Summary (Sheet 13)

* **PTI 78-01** - Spent Fuel Pool Cooling

    Perform pre fuel load testing (Refueling Water Purification Subsystem) as a combination of ATI and/or Component level testing
    Delete Pre-Fuel Load FSAR Test Summary (Sheet 15,16)

* **PTI 62-02** - Boric Acid

    Defer boron operations until just prior to Fuel Load
    Modify Pre-Fuel Load FSAR Test Summaries (Sheets 18,19,21)

* **PTI 84-01** - Flood Mode Boration

    Accomplish using a combination of Work Order (spool piece fitup), ATI and/or component level testing
    Delete FSAR Test Summary (Sheet 20)

* **PTI 261-02 through 05** - Process Computer Tests

    Test via a combination of ATI and/or Component level testing
    Delete FSAR Test Summary (sheet 54)

* **PTI 252-01** - Paging and Evacuation Alarms
* **PTI 256-01** - Sound Powered Primary and Alternate Shutdown Communications

    Perform as in conjunction with Emergency Drill preparations, using either plant procedures, ATI, and/or Component Level testing. It should be noted that phone communications for Remote Shutdown capability were tested during performance of PTI 68-13.
    Delete FSAR Test Summary (Sheet 56)

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FSAR CHANGE REQUEST

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II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
ENGINEERING			
TECH SUPPORT	Landy L. McCormick	Landy L McCormick	6/8/94
OPERATIONS			
LICENSING			

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ~~X~~ or DO NOT ~~X~~ affect the SER. MM 5/6/94

EVALUATED BY: MMBerlin (Name) SUT ENGR (Title) 4/13/94 (Date)

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: MMBerlin (NAME AND SIGNATURE) 4/13/94 (DATE)

APPROVED BY: [signature] M. Bayston 4/24/94 (NAME AND SIGNATURE) 4/13/94 (DATE)

VI. Approved ☐ Rejected ☐
Reason for Rejection: ______

Licensing Engineer: ______ Date: ______ R3

APPENDIX C

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Page 1 of 1

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SUPPORTING ORG	NAME	Signature	Date
ENGINEERING			
TECH SUPPORT			
OPERATIONS			
LICENSING	Charlie Touchstone	CWTouchstone	6-2-94

5/6/94

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☒ or DO NOT ☒ affect the SER.

EVALUATED BY: MMBerlin (Name) SUT ENGR (Title) 4/13/94 (Date)

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: MMBerlin (NAME AND SIGNATURE) 4/13/94 (DATE)

APPROVED BY: RLWalden (NAME AND SIGNATURE) 4/13/94 (DATE)

VI. Approved ☐ Rejected ☐
Reason for Rejection: ______

Licensing Engineer: ______ Date: ______ R3

APPENDIX C

Page 1 of 1

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(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
ENGINEERING	FAKoontz Jr for RHJohnson	[signature]	6-4-94
TECH SUPPORT			
OPERATIONS			
LICENSING			

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☒ or DO NOT ☒ affect the SER. 5/6/94

EVALUATED BY: [signature] (Name) SUT ENGR (Title) 4/13/94 (Date)

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: [signature] NAME AND SIGNATURE 4/13/94 DATE

APPROVED BY: [signature] NAME AND SIGNATURE 4/13/94 DATE

VI. Approved ☐ Rejected ☐
Reason for Rejection: ______

Licensing Engineer: ______ Date: ______ R3

APPENDIX C

Page 1 of 1

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FSAR CHANGE REQUEST

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II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
~~ENGINEERING~~			
~~TECH SUPPORT~~			
OPERATIONS	D S King	D S King	5/2/94
~~LICENSING~~			

MH 5/6/94

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☒ or DO NOT ☒ affect the SER.

EVALUATED BY: MM Berlin (Name) SNT ENGR (Title) 4/13/94 (Date)

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: MM Berlin (NAME AND SIGNATURE) 4/13/94 (DATE)

APPROVED BY: R C Weber (NAME AND SIGNATURE) 4/13/94 (DATE)

VI. Approved ☐ Rejected ☐
Reason for Rejection: ______ R3

Licensing Engineer: ______ Date: ______

FSAR Chapter 6
Reason for Changes By Page

6.3-21 To clarify the level of testing for the Flood Detection System, consistent with the changes being made to chapter 14

FSAR Chapter 14
Reason for Changes By Page

14.2-16 To clarify the type of testing to be performed on boron control systems during the preoperational test phase.

To clarify the intended level of testing to be performed on the Ice Condenser, Seismic Instrumentation, Leak Detection, and Flood Detection Systems.

14.2-17 To clarify the level of testing to be performed on the Process Computer and Fuel Pool Cooling Systems.

14.2-18 To clarify the intended level of testing to be performed on the Communications System, Intake Pump Station Ventilation System, Drain System, and CO2 Fire Suppression System.

14.2-24 To clarify that system operations using boron will not be performed during the preoperational test phase.

TABLE 14.2-1

Sheet

12 Deleted. PTI testing will not be performed, as specified above.

13 CO2 System testing will not be performed by PTI, as specified above.

15 Deleted. PTI testing will not be performed, as specified above.

16 Deleted. PTI testing will not be performed, as specified above.

18 Boron adjustment will not be accomplished during the preoperational test phase, as noted above.

19 Boron system testing during the preoperational test phase will be accomplished using demineralized water, simulating boron operations.

TABLE 14.2-1

Sheet	
20	Deleted. Testing of these non-safety related components will be accomplished by Work Order (for spool piece fitup), and a combination of acceptance test and/or component level testing.
21	Heat Tracing will not be tested as part of a PTI, as noted in the exception to RG 1.68, Appendix A, subparagraph 1.n.18 stated in Chapter 14 of the FSAR.
54	Deleted. PTI testing will not be performed, as specified above.
56	Deleted. PTI testing will not be performed, as specified above.
62	Deleted. PTI testing will not be performed, as specified above.
88	Deleted. PTI testing will not be performed, as specified above.
90	Deleted. PTI testing will not be performed, as specified above.

INSERTS FOR APPENDIX A

A. Appendix A, subparagraph 1.b.2, and 1.n.12

Boration will not be accomplished during preoperational testing. These chemical control system tests will be performed by simulating boron system operations using demineralized water. System operations using boron will be accomplished as part of Surveillance testing in preparation for the Power ascension phase. (File Note: Table 14.2-2 abstracts discuss boron concentration per R. Warren, Rx engr.; its done via SIs) cwt 6/21/94

B. Appendix A, subparagraph 1.h.3

~~Gross Bypass Leakage testing will be performed by PTI.~~ cwT 7/7/94 ~~However, other~~ Testing of ~~this~~ the Ice Condenser system will be accomplished by a combination of acceptance and/or component tests, and /or surveillance testing. Air return fans are tested under Containment Ventilation preoperational tests. (File Note: T/Ss address Ice Condenser TESTING) 7/7/94

C. Appendix A, subparagraph 1.j.7, and 1.j.20

Because of it's non safety related nature, testing of this system will be accomplished by a combination of acceptance tests and/or component level testing.

D. Appendix A, subparagraph 1.j.10

Seismic instrumentation will be calibrated as part of component level testing.

E. Appendix A, subparagraph 1.j.25

Testing of the non-safety related process computer will be accomplished by a combination of acceptance tests and/or component level testing.

F. Appendix A, subparagraph 1.m.1

As new fuel is currently stored in the spent fuel pool, only the refueling water purification subsystem will be tested during the preoperational phase. Because of its non-safety related nature, the testing will be accomplished by a combination of acceptance and/or component level testing. The balance of system testing will be conducted prior to placing irradiated fuel into the spent fuel pit.

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G. Appendix A, subparagraph 1.n.7

Verification that each enclosure which utilizes a CO_2 fire suppression system is provided with appropriate CO_2 concentrations in accordance with design requirements will be performed by either an actual CO_2 discharge or by integration of enclosure air leakage data with previous CO_2 discharge test data. Air leakage data will be obtained by performing a pressurization test for the enclosure and measuring the air leakage to determine the CO_2 retention time. This non-safety related testing will be performed by a combination of acceptance tests and/or component level testing.

H. Appendix A, subparagraph 1.n.9

LICENSING EDITORIAL
AWT 6-21-94

Testing of the (non-safety related) drain system will be accomplished by ~~a combination of~~ acceptance test and/or component level testing.

I. Appendix A, subparagraph 1.n.13

Testing of the non-safety related communications system will be accomplished in conjunction with Emergency Drill preparations, using either plant procedures, acceptance tests or component level testing.

J. Appendix A, subparagraph 1.n.14

Because of its non-safety related nature, testing of the intake pump station ventilation system will be accomplished by a combination of acceptance tests and/or component level testing.

unit (elevation 676), and in the pipe chase (elevation 692). A common alarm in the main control room will alert the operator when any of these flood detectors are tripped. A flood detector indicator panel, located immediately outside the control room, then identifies the exact location of the tripped detector. The detectors are to be ~~preoperationally tested~~ to verify initial operability and will be periodically tested as a part of the plant instrument surveillance and maintenance program.

TESTED DURING THE PREOPERATIONAL TEST PHASE

Since each ECCS pump compartment is monitored by a level detection device, the operator may immediately identify which subsystem must be shut down and secured to terminate the leak.

The operator can readily accomplish this action from the main control room by stopping the appropriate subsystem pump, and by closing the corresponding sump isolation valves and individual pump discharge valves. The time necessary for the operator to detect leakage in a pump compartment is dependent on the leakage rate. A limiting 50 gpm leak in the largest ECCS pump compartment can be detected within 30 minutes. Slower leaks may require proportionally longer detection times.

Leakage into these ECCS pump compartments is piped to the tritiated equipment drain sump. The drain in each of these rooms is provided with a standpipe which assures that the setpoint for the level detector is reached prior to draining the leakage from the room. However, the standpipes each have two 1/2-inch drilled holes to allow minor normal leakage to drain from the room. The tritiated equipment drain sump has redundant 50 gpm pumps which automatically discharge on high level to the tritiated drain collector tank. Operation of these pumps is indicated in the main control room. Additionally, both the tritiated equipment drain sump and collector tank have high level alarms which indicate in the main control room. If the waste disposal system is available, the operator can manually initiate processing of the contents of the tritiated drain collector tank through the waste disposal system. If the waste disposal system is not available, the tritiated drain collector tank will fill and discharge through overflow piping to the Auxiliary Building passive sump.

ECCS leakage into the Auxiliary Building locations other than the ECCS pump compartments is piped to the Auxiliary Building floor and equipment drain sump. This sump is provided with redundant 50 gpm pumps which are indicated in the main control room. The floor drain collector tank is provided with overflow piping which discharges to the Auxiliary Building passive sump. Leakage into these areas can be detected by the flood detection system described above, by indication of sump pump operation, or by high level alarm from the sump or the floor drain collector tank. However, the exact location of the leak, if other than the ECCS pump compartment, or the subsystem from which leakage occurs, may not be immediately identified. Since ECCS leaks other than a pump seal failure are of a nature to develop very slowly and are less severe than a seal failure, the operator has an extended time period to detect and isolate the leak. Isolation of these minor leaks can be accomplished by arbitrarily selecting and isolating an ECCS subsystem and evaluating the response of the flood detector system.

test and determined that additional testing of welds and weld repairs is necessary. This additional testing, as well as a leak test of the RCS in accordance with ASME Section XI, will be performed as described in References [1] and [2] using appropriate test procedures. The results of these tests will be approved by JTG and retained as plant records in accordance with FSAR Section 14.2.6.

INSERT A

(b) Appendix A, subparagraph 1.c

Acceptance criteria for the response time of the various logic channels will be consistent with Technical Specifications requirements. The Reactor Trip System Response Time is defined in the Technical Specifications as the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The accident analysis accounts for conservative values for delay times, setpoint drift, etc. Therefore, it is not necessary to account for the response time of the associated hardware between the measured variable and the input to the sensor in the test or acceptance criteria.

(c) Appendix A, subparagraph 1.g.2

Emergency loads will not be tested with minimum and maximum design voltage available. Emergency loads will be tested to demonstrate satisfactory starting and operating characteristics with power supply voltage within the design operating range. Transformer taps will be adjusted to obtain optimum voltage levels from no-load to full load conditions. Tests will be performed to record operating parameters of the offsite grid, Class 1E 6.9 kV, 480 volt, and 120 volt vital power busses under no-load, steady state load, and transient conditions. Data will be obtained for the Class 1E train having the lowest analyzed voltage. The recorded information will be compared to engineering voltage calculations to validate the analytical models used.

INSERT B, C & D

(d) Appendix A, subparagraph 1.j.12

The gross failed fuel detection system (GFFDS) will not be tested for Unit 1 operation. As discussed in Section 9.3.5, the GFFDS is installed at WBN but no credit is taken for its use in identifying conditions of fuel failure. The GFFDS performs no safety related function and is not designed to satisfy any specific safety criteria. Periodic sampling is used to detect failed fuel as described in Section 9.3.2.2

10 CFR 50, Appendix A and RG 1.97 do not require a GFFDS and, from a practical standpoint, there is no justification for the monitor. This particular design detects delayed neutrons and any type of failure that would be detected by this monitor would be putting significant fuel particles into the coolant.

To damage fuel to this extent would require either a severe transient, foreign material damage or an unusual design problem, such as baffle jet impingement. The plant is designed to operate such that clad damage will not occur. Technical Specifications limits on pressure, temperature, flow and power ensure that cladding is not damaged during normal operations, including anticipated transients. Reactor coolant is sampled frequently. Although Technical Specifications only require gross specific activity and dose equivalent I-131 analyses every 7 days and 14 days during power operations, coolant will actually be sampled for other parameters at least every 72 hours. The radcon surveys required to handle these samples would detect any significant activity changes. If power changes 15% or more in any one hour, a special sample is required between 2 and 6 hours after the power change.

Due to the ability to monitor the conditions that damage fuel cladding, it is easily determined if sampling should be performed for evidence of damage. Monitors for subcooling margin, incore thermocouples, and reactor vessel level are examples of monitors that would indicate clad damaging conditions.

Foreign material damage (monitored by noise monitor) and baffle jet impingement have resulted in non-catastrophic clad failures at other plants which were detected and monitored by normal chemistry sampling.

(e) Appendix A, subparagraphs 1.j.22, 1.k.2, and 1.k.3

The subject equipment is calibrated and functionally tested as part of the WBN plant instrument calibration program. The calibration and functional testing is performed and documented in accordance with approved plant calibration procedures. Therefore, additional testing in the form of a preoperational test is not warranted.

→ INSERT E

(f) Appendix A, subparagraph 1.k.4

Refer to 14.2.7.3 for discussion of compliance with requirements for filter and in-place leak tests.

(g) Appendix A, subparagraph 1.m.1

→ INSERT F

~~As fuel is currently stored in the spent fuel pool, only the refueling water purification subsystem will be tested prior to fuel load. This exception will be processed as described in Section 14.2.1.~~

(h) Appendix A, subparagraph 1.m.4

Static load testing at 125% of rated load for equipment and components used to handle irradiated or non-irradiated fuel originates from an ANSI B30.2 rated load test requirement. This test is purposed to verify the structural integrity of the handling equipment and is utilized to rate its capacity. Handling equipment is rated at 80% of the test load, resulting in a rating of 100% capacity when test loads

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are 125%. This ANSI testing is required to be performed prior to initial use, and following extensive repairs or modifications. Three of the four Unit 1 handling devices utilized to handle fuel (Spent Fuel Pit Bridge Crane, Refueling Machine, and 125T Auxiliary Building Crane main hook), as well as both hooks of the Polar Crane, have been successfully load tested at 125% rated capacity in prior tests. Additionally, this equipment has had no extensive maintenance or modifications which would affect structural integrity. Therefore, repeated load testing of this equipment is not warranted. The fourth device, the auxiliary hook of the 125T Auxiliary Building Crane, handles new fuel but has not been load tested at 125% capacity. Therefore it will be load tested to 125% of its ten-ton capacity.

→ INSERT G, H, I & J

(i) Appendix A, subparagraph 1.n.18

Because of its non-safety related, and simple functions, trace heating systems will receive component level tests, not preoperational or acceptance testing.

(j) Appendix A, subparagraph 2.b

Cold no flow, cold full flow and hot no flow rod drops do not provide any additional useful data. The drop times for these flow conditions are less conservative than for hot full flow conditions. We do not intend to perform cold no flow, cold full flow and hot no flow rod drops.

(k) Appendix A, subparagraphs 2.f and 5.m

WBN does not intend to perform a differential pressure measurement across the core nor a core flow measurement since these are prototype tests. WBN is not a prototype model plant, but, instead, is a well documented production model with several similar predecessor reactor units (Indian Point Unit 2, Trojan, and Sequoyah Unit 1) of its type. Proper differential pressures across the core and core flow is shown indirectly through performance of other tests that verify operating temperature and RCS flow.

(l) Appendix A, subparagraphs 4.c and 5.e

No new design information is to be gained from the RCCA Pseudo Ejection Test. WBN is not a prototype plant for 4 loop, 12 foot core with 17 x 17 fuel design and B_4C control rods. Performance and measurement data already exists for this design. Therefore, WBN will not perform this test.

(m) Appendix A, subparagraphs 4.q and 5.z

The proper response of process and effluent radiation monitors is demonstrated under the plant calibration program and during preoperational testing for the Process and Effluent Radiation Monitoring System (Sheet 31, Table 14.2-1). It is not expected that enough leakage in fuel cladding, steam generator tubes or heat exchangers will exist to provide a meaningful comparison between radiation monitor responses and laboratory analysis of samples.

1079 S00 PKG

(f) Appendix A, subparagraph 5.m

Sufficient measurements and evaluations should be conducted with the plant at steady-state conditions to establish that vibration levels of reactor coolant system components are in agreement with design values.

(g) Appendix A, subparagraph 5.r

Verify by review and evaluation of printouts and/or cathode ray tube (CRT) displays that the control room or process computer is receiving correct inputs from process variables and validate that performance calculations performed by the computer are correct.

(h) Appendix A, subparagraph 5.s

Calibrate, as necessary, and verify the performance of major or principle plant control systems, including boron addition systems (PWR), main, auxiliary and emergency feedwater control systems; hotwell level control systems; and reactor coolant makeup and letdown control systems.

[EXCEPT FOR ACTUAL BORON MIXING OR MOVEMENT]

Use Brackets CCT 6/21/94

(i) Appendix A, subparagraph 5.y

Calibrate, as required, and verify the proper operation of important instrumentation systems including reactor coolant level.

(j) Appendix A, subparagraph 5.d.d

Demonstrate the potential capability for placing the reactor in a cold shutdown condition.

(k) Appendix A, subparagraph 5.e.e

Demonstrate that primary containment purge system operates in accordance with design.

(5) Some provisions of RG 1.68 are satisfied at other plant conditions than specified. These include:

(a) Appendix A, subparagraph 4.e

Determination made up to 30% power rather than from 0 to 5% power.

(b) Appendix A, subparagraph 4.i

Demonstration done in mode 5 and mode 3 rather than from 0 to 5% power.

(c) Appendix A, subparagraph 4.o

Testing performed in mode 3 rather than from 0 to 5% power.

TABLE 14.2-1

(Sheet 12 of 90)

LIQUID WASTE DRAINS, COLLECTION AND TRANSFER SYSTEM
TEST SUMMARY

OBJECTIVE

Demonstrate the capability of the floor and equipment drains to direct drainage from areas housing safety related equipment, radioactive and potentially radioactive liquids, chemicals and oils to designated collection points for transfer to storage tanks or processing systems.

PREREQUISITES

1. Floor and equipment drain lines and sumps have been cleaned of construction debris and are capable of receiving and transferring liquids.

2. Drain collector tanks and associated transfer pumps are operable.

3. AC and DC electrical power supplies are available.

TEST METHOD

1. Deliver water to each floor and equipment drain and verify capability of the drains to remove the water.

2. Verify setpoints for sump levels and pump actuation.

3. Verify flood detection instrumentation and alarm actuation.

4. Verify ECCS room passive sump and alarm actuation.

ACCEPTANCE CRITERIA

1. All floor and equipment drains are clear of obstruction and direct waste liquids to the proper location.

2. RHR and Containment Spray Pump compartment drains transfer liquid waste at design flowrates as described in FSAR Section 9.3.3.

3. Automatic controls, interlocks and alarms operate in accordance with design drawings.

4. Sump and/or drain pumps operate in accordance with design drawings to control sump or tank level as described in FSAR Section 9.3.3.

5. Flood detection instrumentation and alarm operates as described in FSAR Section 6.3.2.11.3.

DELETED

1079 S00 PKG

TABLE 14.2-1

(Sheet 13 of 90)

FIRE PROTECTION SYSTEM
TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the Fire Protection System, including its fire detection and fire suppression functions, in accordance with design requirements.

PREREQUISITES

1. The Fire Protection System pumps, piping, controls and associated valves and dampers are operational.

2. Fire protection and detection instrumentation is calibrated.

TEST METHOD

1. Verify the proper functioning of the fire detection devices to activate the automatic fire protection system, alert the appropriate control location, initiate fire alarms, and to activate automatic closure of fire dampers, as required.

2. Verify proper operation of the fire suppression system, and obtain flow rates through the underground loop and differential pressure across the strainers.

3. Demonstrate the automatic start feature of the Fire Protection System pumps.

4. Demonstrate the capability of the Fire Protection System and pumps to supply water to required areas of the plant at design flow and pressure.

5. Verify system vibrations are within design limits.

6. Demonstrate proper operation of system instrumentation, alarms, controls and interlocks.

7. Verify the Aqueous Film Forming Foam system proportioning equipment operates in accordance with design and vendor documents.

8. Verify that each enclosure which utilizes a CO_2 fire suppression system is provided with appropriate CO_2 concentrations in accordance with design requirements. The test will be performed by either an actual CO_2 discharge or by integration of enclosure air leakage data with previous CO_2 discharge test data. Air leakage data will be obtained by performing a pressurization test for the enclosure and measuring the air leakage to determine the CO_2 retention time.

TABLE 14.2-1

(Sheet 15 of 90)

SPENT FUEL POOL COOLING SYSTEM
(Refueling Water Purification Portion)
TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Refueling Water Purification Subsystem of the Spent Fuel Pool Cooling System to provide required water flows to the refueling cavity, RWST, and transfer canal and verify proper operation of the purification loop.

PREREQUISITES

1. The refueling cavity, transfer canal, and RWST are filled with demineralized or borated water as required for portions of the test.

2. Spent fuel pool cooling demineralizer has been loaded as required for portions of the test.

3. Refueling water purification filters are installed as required for portions of the test.

TEST METHOD

1. Verify proper operation and actuation of pumps and valves in all operational modes and verify correct flows in the refueling water purification loops including:

   - the ability to transfer water to and from the transfer canal and refueling cavity.

   - the recirculation capabilities using the refueling water purification pumps and verify flow through the refueling water purification filters and spent fuel pit demineralizer.

2. Check operation of instrumentation, interlocks, and alarms for the Spent Fuel Pool Cooling System .

3. Verify no vortexing occurs during various modes of operation.

4. Install prefabricated spool pieces for the Spent Fuel Pool Cooling System as required for flood mode operation.

DELETED

TABLE 14.2-1

SPENT FUEL POOL COOLING SYSTEM
(Refueling Water Purification Portion)
TEST SUMMARY

ACCEPTANCE CRITERIA

1. The hydraulic performance of the refueling water purification pumps meets or exceeds the design requirements as described in FSAR Section 9.1.3.

2. Automatic and manual controls, interlocks, and alarms operate in accordance with design documents.

3. All system flood mode preparations can be made in accordance with design requirements as described in FSAR Section 2.4.14.

DELETED

1079 S00 PKG

TABLE 14.2-1

(Sheet 18 of 90)

CHEMICAL AND VOLUME CONTROL SYSTEM
TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the Chemical and Volume Control System (CVCS), including the capability to maintain charging and letdown flows, to maintain seal-water injection flow to the reactor coolant pumps, to maintain water chemistry conditions, to provide reactor makeup control~~, and to adjust reactor coolant boron concentration.~~

AND

PREREQUISITES

1. Applicable portions of the Reactor Coolant System are capable of operationally interfacing with the Chemical and Volume Control System as required.

2. A cooling water supply is available for the heat exchangers as necessary for test performance.

3. Systems required to supply cover gas to the Volume Control Tank (VCT) are operational, and adequate supplies of gas are available.

TEST METHOD

1. Verify proper functioning of charging and letdown system components, including the hydraulic performance of the charging pumps, and operability of heat exchangers, letdown orifices, and control valves.

2. Demonstrate the capability to maintain seal water flow to the reactor coolant pumps.

3. Verify proper flows and pressure drops for seal injection and reactor coolant filters.

4. Verify proper operation of the volume control tank level and pressure control, including testing of the automatic makeup, dilution, alternate dilute, borate, and manual modes of reactor makeup control and cover gas system.

5. Check proper operation of instrumentation, interlocks, and alarms.

6. Verify proper operation of pumps and valves in the boric acid subsystem.

7. Demonstrate the chemical control function of the CVCS by verifying the capability of the system to introduce chemicals into the charging flow for pH and oxygen control, and that the system is capable of maintaining a gas pressure in the volume control tank as required during the applicable modes of operation.

1079 S00 PKG

TABLE 14.2-1

(Sheet 19 of 90)

CHEMICAL AND VOLUME CONTROL SYSTEM
TEST SUMMARY

TEST METHOD (Cont'd)

8. Verify proper flows to the in-service mixed bed and the cation demineralizer, and determine pressure drops and effectiveness of demineralizers and filters.

BY SIMULATION USING DEMINERALIZED WATER,

9. Verify that a solution of boric acid ~~can~~ be mixed, transferred, recirculated, and stored. ~~COULD~~ STET *

10. Verify that boric acid ~~can~~ be transferred to other systems as required. ~~COULD~~ STET *

ACCEPTANCE CRITERIA

1. The hydraulic performance of the charging pumps meets or exceeds design requirements as described in FSAR Section 9.3.4.

2. Charging and letdown normal and alternate flowpaths, including heat exchangers, letdown orifices, and control valves, operate in accordance with design requirements as described in FSAR Section 9.3.4.

3. Automatic and manual controls, including chemical and automatic reactor makeup water control, interlocks, and alarms, operate in accordance with design drawings.

AS DEMONSTRATED BY SIMULATION USING DEMINERALIZED WATER,

4. Boric acid ~~can~~ be batched, stored and transferred in accordance with design requirements as described in FSAR Section 9.3.4.

~~COULD~~ STET * CWT 6/21/94 (licensing editorial)

1079 S00 PKG

TABLE 14.2-1

(Sheet 20 of 90)

FLOOD MODE BORATION SYSTEM
TEST SUMMARY

OBJECTIVE

To demonstrate that the Auxiliary Charging equipment is capable of supplying makeup water flow to the Reactor Coolant System as required during flood mode operations.

PREREQUISITES

1. All necessary support systems are operational as required.
2. All equipment required for flood mode preparation is available.
3. Auxiliary Charging filters are installed and demineralizer loaded.

TEST METHOD

1. Verify proper operation of instrumentation, alarms and interlocks.
2. Demonstrate the equipment's ability to provide required flow to the Reactor Coolant System from the various makeup water sources.
3. Install all equipment required for flood mode operation.
4. Transfer makeup water from the preferred sources to the auxiliary makeup tank.
5. Connect fire hose between the High Pressure Fire Protection System and the auxiliary makeup tank. (Fire water is not to be introduced into the tank).
6. Verify proper operation of the auxiliary boration tank mixer.

ACCEPTANCE CRITERIA

1. The hydraulic performance of the auxiliary charging pumps and booster pumps meets or exceeds design requirements as described in FSAR Section 9.3.6.
2. Equipment required for flood mode operation can be properly installed.
3. All automatic controls, interlocks and alarms function in accordance with design drawings.

DELETED

TABLE 14.2-1

BORON RECYCLE SYSTEM
TEST SUMMARY

OBJECTIVE

Demonstrate the operability of the Boron Recycle System.

PREREQUISITES

1. The recycle evaporator of the Boron Recycle System, and other interrelated or supporting equipment, are operational as required (Unit 2).

2. A steam supply and cooling water supply are available for the evaporator packages as required (Unit 2).

TEST METHOD

1. Demonstrate the capability of the Boron Recycle System to collect water from its various designated sources (Unit 1 and 2).

2. Verify proper operation of system components, including the evaporator package and the feed, distillate and concentrates pumps (Unit 1 and 2, as appropriate).

3. Verify the proper operation of the piping heat tracing (Unit 2).

4. Check for operability of alarms and interlocks (Unit 1 and 2, as appropriate).
3

ACCEPTANCE CRITERIA

1. The Boron Recycle System components function in accordance with design and vendor documents as described in FSAR Section 9.3.7 (Unit 1 and 2, as appropriate).

2. Flow paths are verified operational (Unit 1 and 2).

3. Heat tracing and other system related heating methods maintain system minimum temperatures in accordance with design documents (Unit 2).

4. Automatic controls, interlocks, and alarms operate properly in accordance with design drawings and vendor documents (Unit 1 and 2, as appropriate).
3

TABLE 14.2-1

COMPUTER SYSTEM
TEST SUMMARY

OBJECTIVE

The purpose of this test is to verify the P2500 process computer has been internally wired properly and that the internal CPU, I/O and analog converters function properly.

To verify the operation of the computer for conversion and computer printout of process parameters.

To verify the proper operation of the P2500 computer software.

PREREQUISITES

1. The latest Operating System Software and Data files are loaded and operational.

2. The proper signal conditioners have been installed on the computer half shells.

TEST METHOD

1. Verify the software functions are processed accurately by the hardware.

2. Verify that the control processing and peripheral hardware operates in a manner to satisfy all requirements. (Note: Loop calibration is performed in the component test program.)

3. Verify the P2500 computer will appropriately process input signals.

ACCEPTANCE CRITERIA

1. The P2500 Process Computer has been internally wired properly and the internal CPU, peripheral devices, Input/Output (I/O) and analog converters function properly.

2. The calibration and operation of the elements of the P2500 Plant Computer results in accurate processing and display of analog and digital input signals using the P2500 Plant Computer System.

3. Installed application programs perform as designed.

DELETED

TABLE 14.2-1

(Sheet 56 of 90)

COMMUNICATIONS SYSTEM
TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Sound-Powered Telephone System and the Codes, Alarms and Paging (CAP) System to provide adequate communication coverage and audibility.

PREREQUISITES

1. Installation and construction testing of the Plant Communication and Evacuation Alarm Systems.

2. Plant equipment and systems are operating so as to create an ambient noise level that would be expected during normal plant operation.

TEST METHOD

1. Demonstrate proper functioning of the Sound-Powered Telephone System to provide intelligible reception and transmission of voice communications between assigned locations.

2. Verify proper functioning and audibility of the CAP System to provide paging and sound evacuation, fire, and medical alarms.

ACCEPTANCE CRITERIA

1. The codes, alarms, and paging system provides for audible paging and alarm signaling as described in FSAR Section 9.5.2.

2. The sound powered telephone system provides for audible communication between locations described in FSAR Section 9.5.2.

3. Evacuation, fire, and medical alarms, all clear signals and personnel pages are audible with normally expected background noise level.

DELETED

1079 S00 PKG

TABLE 14.2-1

SEISMIC INSTRUMENTATION
TEST SUMMARY

OBJECTIVE

To demonstrate the operability of installed seismic instrumentation and its capability to monitor and record.

PREREQUISITES

1. A calibrated seismic test signal is available as required by the vendor manuals.

TEST METHOD

1. Verify that the seismic test signal will activate the instrumentation as required at an acceptable level.

2. Demonstrate proper operation of seismic instrumentation components and ~~alarms~~, including the triggering device, recording and playback system, and peak acceleration recorders.

ACCEPTANCE CRITERIA

1. The seismic instrumentation is capable of being aligned in accordance with the vendor technical manual.

2. The system instruments, including trip settings, demonstrate correct response and outputs in response to simulated input signals as described in FSAR Section 3.7.4.

DELETED

1079 S.v 0 P.K G

TABLE 14.2-1

(Sheet 88 of 90)

ICE CONDENSER SYSTEM.
TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the Ice Condenser Doors, associated air handling components, glycol circulation, refrigeration, floor cooling, and drain subsystems.

PREREQUISITES

1. All system components have been installed.

2. The Raw Cooling Water System and Demineralized Water System are available.

3. System relief valve setpoints have been verified.

4. Heat tracing is installed on Air Handling Unit drains and is available for use.

5. Air handling units, glycol circulation and refrigeration equipment is operable.

TEST METHOD

1. Demonstrate the proper operation of the refrigeration chiller packages, glycol circulation equipment, air handling units and the floor cooling and drain system.

2. Demonstrate the ice condenser can be adequately cooled to and maintained at design conditions.

3. Verify the ice condenser has been loaded with the proper quantity and quality of the ice.

4. Verify all system controls, interlocks, instrumentation, and alarms function properly to simulated or actual signals.

ACCEPTANCE CRITERIA

1. The Ice Condenser System components, controls, interlocks, instrumentation and alarms function in accordance with FSAR Section 6.7 and the appropriate design basis documents.

2. The Ice Condenser has been loaded with the proper quantity and quality of borated ice as required in FSAR Section 6.7.

DELETED

TABLE 14.2-1

INTAKE PUMP STATION VENTILATION SYSTEM
TEST SUMMARY

OBJECTIVE

Demonstrate proper operation of the Intake Pump Station (IPS) ventilation equipment.

PREREQUISITES

1. System sufficiently complete to support testing.

2. AC electrical power supplies are available.

TEST METHOD

1. Demonstrate proper operation of the ventilation supply and exhaust fans, shutoff louvers, and dampers.

2. Verify all alarms and interlocks function properly.

3. Verify proper operation of the IPS unit and duct heaters.

ACCEPTANCE CRITERIA

1. Manual and automatic controls, interlocks, auto start features, and alarms function in accordance with design documents.

DELETED

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APPENDIX C
Page 1 of 1

FSAR CHANGE REQUEST

FSAR CHANGE PACKAGE NO. 1 0 8 0 ______ Supp. No.

I. Attached are proposed changes to FSAR pages with justification for each change:

SECTION, PAGE NO. FIGURE OR TABLE	REASON FOR CHANGE
1) SEE ATTACHED	
2)	
3)	
4)	
5)	
6)	
7)	
8)	
9)	
10)	

(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations.

SUPPORTING ORG	NAME	SIGNATURE	DATE
ENGINEERING			
TECH SUPPORT	Landy L. McCormick	[signature]	6/8/94
OPERATIONS			
LICENSING			

CRAIG PARKER PER TEL 6/27/94

[signature] 6/27/94

III. The proposed changes have been evaluated with respect to potential effects on conclusion drawn in the Safety Evaluation Report (SER). Changes do ☐ or DO NOT ☒ affect the SER.

EVALUATED BY: [signature] (Name) SUT ENGINEER (Title) 4/20/94 (Date)

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: [signature] (Name and Signature) 4/20/94 (Date)

APPROVED BY: [signature] (Name and Signature) 4/20/94 (Date)

VI. Approved ☐ Rejected ☐ Amendment No. ______

Reason for Rejection: ______

Licensing Engineer: ______ Date: ______

WBN 0	MANAGEMENT OF THE FINAL SAFETY ANALYSIS REPORT (FSAR)	SSP-4.02 Revision 4 Page 26 of 27

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(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations.

SUPPORTING ORG	NAME	SIGNATURE	DATE
ENGINEERING			
TECH SUPPORT			
OPERATIONS	David S. King	D S King	3/22/94
LICENSING			

III. The proposed changes have been evaluated with respect to potential effects on conclusion drawn in the Safety Evaluation Report (SER). Changes do ☐ or DO NOT ☒ affect the SER.

EVALUATED BY: [signature] (Name) SUT ENGINEER (Title) 4/20/94 (Date)

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: [signature] Name and Signature 4/20/94 Date

APPROVED BY: [signature] Name and Signature 4/22/94 Date

VI. Approved ☐ Rejected ☐ Amendment No. ______

Reason for Rejection: ______

Licensing Engineer: ______ Date: ______

APPENDIX C
Page 1 of 1

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(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations.

SUPPORTING ORG	NAME	SIGNATURE	DATE
ENGINEERING	EA Koontz Jr for RMJ	[signature]	6-20-94
TECH SUPPORT			
OPERATIONS			
LICENSING			

III. The proposed changes have been evaluated with respect to potential effects on conclusion drawn in the Safety Evaluation Report (SER). Changes do ☐ or DO NOT ☒ affect the SER.

EVALUATED BY: [signature] (Name) SUT ENGINEER (Title) 4/20/94 (Date)

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: [signature] (Name and Signature) 4/20/94 (Date)

APPROVED BY: [signature] (Name and Signature) 4/20/94 (Date)

VI. Approved ☐ Rejected ☐ Amendment No. ______

Reason for Rejection: ______

Licensing Engineer: ______ Date: ______

1080 S00 PKG

WBN 0	MANAGEMENT OF THE FINAL SAFETY ANALYSIS REPORT (FSAR)	SSP-4.02 Revision 4 Page 26 of 27

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10)	

(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations.

SUPPORTING ORG	NAME	SIGNATURE	DATE
~~ENGINEERING~~			
~~TECH SUPPORT~~			
~~OPERATIONS~~			
~~LICENSING~~ TW	CW Touchstone	CW Touchstone	6-7-94 ~~w/ comment sheet~~ 75/70 OK CWT 6/21/94

III. The proposed changes have been evaluated with respect to potential effects on conclusion drawn in the Safety Evaluation Report (SER). Changes do ~~X~~ or DO NOT ~~X~~ affect the SER. CWT 6/21/94 CWT 6/21/94

EVALUATED BY: [signature] Name SCT ENGINEER Title 4/20/94 Date

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: [signature] Name and Signature 4/20/94 Date

APPROVED BY: [signature] Name and Signature M. Bajestani 6/24/94 4/20/94 Date

VI. Approved ☐ Rejected ☐ Amendment No. ____________

Reason for Rejection: ____________

Licensing Engineer: ____________ Date: ____________

CWT 6/21/94

FSAR Chapter 14
Reason for Changes By Page

14.2-17 ~~17~~ To clarify the intended level of testing to be performed on the Solid Waste Processing System.

14.2-18 To clarify the intended level of testing for accomplishment of crane load tests, and operational tests.

TABLE 14.2-1

Sheet

29 Deleted. PTI Testing will not be performed, as specified above.

74 Crane Load testing and operational tests for cranes not associated with spent fuel movement will not be performed by PTI, as specified above.

75 Same as 74

INSERT FOR APPENDIX A

A. Appendix A, subparagraph 1.1.3

Testing of the non-safety related Solid Waste Processing System will be performed by a combination of Acceptance Tests and/or Component level testing.

B. Appendix A, subparagraph 1.m.4, add to existing wording:

Load testing will be accomplished by Work Order. Operational, Load testing of cranes not associated with spent fuel movement will be accomplished by a combination of Acceptance Tests and/or Component level testing. THE ~~[illegible]~~ OPERATIONAL TEST WILL BE PERFORMED UTILIZING A DUMMY FUEL ASSEMBLY AND THE APPROPRIATE HANDLING TOOL(S), WHICH IS THE OPERATIONAL LOAD THE CRANES WILL SEE DURING FUEL HANDLING.

insert: for the manipulator crane and the spent fuel pit bridge crane

Licensing change made subsequent to approvals to be consistent w/ test abstract - does not change intent of wording. CWT 6/21/94

To damage fuel to this extent would require either a severe transient, foreign material damage or an unusual design problem, such as baffle jet impingement. The plant is designed to operate such that clad damage will not occur. Technical Specifications limits on pressure, temperature, flow and power ensure that cladding is not damaged during normal operations, including anticipated transients. Reactor coolant is sampled frequently. Although Technical Specifications only require gross specific activity and dose equivalent I-131 analyses every 7 days and 14 days during power operations, coolant will actually be sampled for other parameters at least every 72 hours. The radcon surveys required to handle these samples would detect any significant activity changes. If power changes 15% or more in any one hour, a special sample is required between 2 and 6 hours after the power change.

Due to the ability to monitor the conditions that damage fuel cladding, it is easily determined if sampling should be performed for evidence of damage. Monitors for subcooling margin, incore thermocouples, and reactor vessel level are examples of monitors that would indicate clad damaging conditions.

Foreign material damage (monitored by noise monitor) and baffle jet impingement have resulted in non-catastrophic clad failures at other plants which were detected and monitored by normal chemistry sampling.

(e) Appendix A, subparagraphs 1.j.22, 1.k.2, and 1.k.3

The subject equipment is calibrated and functionally tested as part of the WBN plant instrument calibration program. The calibration and functional testing is performed and documented in accordance with approved plant calibration procedures. Therefore, additional testing in the form of a preoperational test is not warranted.

(f) Appendix A, subparagraph 1.k.4

Refer to 14.2.7.3 for discussion of compliance with requirements for filter and in-place leak tests.

INSERT A

(g) Appendix A, subparagraph 1.m.1

As fuel is currently stored in the spent fuel pool, only the refueling water purification subsystem will be tested prior to fuel load. This exception will be processed as described in Section 14.2.1.

(h) Appendix A, subparagraph 1.m.4

Static load testing at 125% of rated load for equipment and components used to handle irradiated or non-irradiated fuel originates from an ANSI B30.2 rated load test requirement. This test is purposed to verify the structural integrity of the handling equipment and is utilized to rate its capacity. Handling equipment is rated at 80% of the test load, resulting in a rating of 100% capacity when test loads are 125%. This ANSI testing is required to be performed prior to initial use, and following extensive repairs or modifications. Three of the four Unit 1 handling devices utilized to handle fuel (Spent Fuel Pit Bridge Crane, Refueling Machine, and 125T Auxiliary Building Crane main hook), as well as both hooks of the Polar Crane, have been successfully load tested at 125% rated capacity in prior tests. Additionally, this equipment has had no extensive maintenance or modifications which would affect structural integrity. Therefore, repeated load testing of this equipment is not warranted. The fourth device, the auxiliary hook of the 125T Auxiliary Building Crane, handles new fuel but has not been load tested at 125% capacity. Therefore it will be load tested to 125% of its ten-ton capacity.

INSERT B

(i) Appendix A, subparagraph 1.n.18

Because of its non-safety related, and simple functions, trace heating systems will receive component level tests, not preoperational or acceptance testing.

(j) Appendix A, subparagraph 2.b

Cold no flow, cold full flow and hot no flow rod drops do not provide any additional useful data. The drop times for these flow conditions are less conservative than for hot full flow conditions. We do not intend to perform cold no flow, cold full flow and hot no flow rod drops.

(k) Appendix A, subparagraphs 2.f and 5.m

WBN does not intend to perform a differential pressure measurement across the core nor a core flow measurement since these are prototype tests. WBN is not a prototype model plant, but, instead, is a well documented production model with several similar predecessor reactor units (Indian Point Unit 2, Trojan, and Sequoyah Unit 1) of its type. Proper differential pressures across the core and core flow is shown indirectly through performance of other tests that verify operating temperature and RCS flow.

(l) Appendix A, subparagraphs 4.c and 5.e

No new design information is to be gained from the RCCA Pseudo Ejection Test. WBN is not a prototype plant for 4 loop, 12 foot core with 17 x 17 fuel design and B_4C control rods. Performance and measurement data already exists for this design. Therefore, WBN will not perform this test.

(m) Appendix A, subparagraphs 4.q and 5.z

The proper response of process and effluent radiation monitors is demonstrated under the plant calibration program and during preoperational testing for the Process and Effluent Radiation Monitoring System (Sheet 31, Table 14.2-1). It is not expected that enough leakage in fuel cladding, steam generator tubes or heat exchangers will exist to provide a meaningful comparison between radiation monitor responses and laboratory analysis of samples.

TABLE 14.2-1 1080 S00 PKG

(Sheet 29 of 90)

SOLID WASTE PROCESSING SYSTEM
TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the solid waste systems to collect and prepare disposable dry active and wet active wastes.

PREREQUISITES

1. All necessary supporting systems are operational.

2. Associated pumps, piping, and controls are operable.

TEST METHOD

1. Demonstrate the ability to receive, hold and transfer spent resins from their source to the bulk disposal outlet.

2. Verify proper operation of associated instrumentation, interlocks and alarms.

ACCEPTANCE CRITERIA

1. Wet active waste can be collected, transferred, and dewatered for shipment in accordance with design requirements as described in FSAR Section 11.5.

2. Dry active wastes can be compacted for shipment in accordance with design requirements as described in FSAR Section 11.5.

3. Automatic and manual controls, interlocks, and alarms operate in accordance with design drawings.

DELETED

TABLE 14.2-1 1080 S00 PKG

FUEL HANDLING ~~AND VESSEL SERVICING EQUIPMENT~~

TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the fuel handling ~~and vessel servicing~~ equipment, including the handling tools and equipment, cranes and fuel transfer system.

To provide for final indexing of the manipulator crane and to establish reference marks for the manipulator crane bridge using a verified dimensionally correct dummy assembly.

To provide the opportunity for training fuel handlers prior to actual fuel loading.

PREREQUISITES

1. The refueling cavity, refueling canal and spent fuel pool are clean and areas adjacent to the system equipment ~~are clear~~.

2. Dummy assembly, test weights and test fixtures are available as required for testing the manipulator crane and spent fuel bridge crane.

3. Load testing of the reactor head and internals lifting fixtures has been completed.

TEST METHOD

1. With the use of a dummy assembly demonstrate the proper operation of all system components, including the manipulator crane, spent fuel pit bridge and electric hoist, new fuel elevator, fuel transfer system, rod ~~control~~ cluster control changing fixture~~s~~, various handling tools, and indexing of the system.

FOR THE EQUIPMENT LISTED IN TEST METHOD 1

2. Verify operation of interlocks and proper setting of limit switches.

3. Demonstrate proper operation of crane and hoist controls including overspeed, overloads and travel limits, and warning devices.

4. Demonstrate hoist, bridge and trolley travel is acceptable.

5. Verify the operation of the hoist braking systems.

6. Perform on ~~a 125% static load test and a 100% full load~~ operational load with a dummy assembly test on the Manipulator Crane and the Spent Fuel Pit Bridge Crane~~, Reactor Building Polar Crane and the Auxiliary Building Overhead Crane~~.

1080 S00 PKG

TABLE 14.2-1

FUEL HANDLING ~~AND VESSEL SERVICING EQUIPMENT~~
TEST SUMMARY

ACCEPTANCE CRITERIA

1. Fuel transfer system operation, controls and interlocks function in accordance with FSAR Section 9.1.4 and the vendor technical manual and drawings.

2. All Manipulator Crane operation, controls and interlocks function in accordance with FSAR Section 9.1.4 and the vendor technical manual and drawings.

3. The RCC change fixture's function to remove, store, and load RCC's has been successfully demonstrated in accordance with FSAR Section 9.1.4.

4. Final Manipulator Crane indexing is completed and the Manipulator Crane has proven repeatability at various core locations.

5. Spent Fuel Pit Bridge, ~~Reactor Building Polar and Auxiliary Building overhead~~ crane controls and interlocks function in accordance with the appropriate design drawings and technical manuals.

6. Manipulator Crane AND the Spent Fuel Pit Bridge Crane~~, Reactor Building Polar Crane and the Auxiliary Building Overhead Crane~~ have been successfully load tested.

APPENDIX C

Page 1 of 1

1089 S00 PKG

FSAR CHANGE REQUEST

FSAR CHANGE PACKAGE NO. 1 0 8 9 ______ Supp. No. R3

I. Attached are proposed changes to FSAR pages with justification for each change:

SECTION, PAGE NO. FIGURE OR TABLE	REASON FOR CHANGE
1) FSAR Sect. 14, pg. 14.2-18	Pg. 14.2-18 corrects item (m) to reflect proper section of R.G. 1.68.
2) Table 14.2-2 sheet 10	Table 14.2-2, sheets 10, 17, 20 & 33 have
3) Table 14.2-2 sheet 17	been changed to reflect plant design,
4) Table 14.2-2 sheet 20	change in testing methodology and
5) Table 14.2-2 sheet 33	clarification of test abstracts.
6)	
7)	
8)	
9)	
10)	

(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
OPS	R.G. MENDE	[signature]	5-16-94
Tech Support	D. L. KOEHL	[signature]	5-19-94
~~Engineering~~			
~~Licensing~~			

JSW 5/19/94

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☒ or DO NOT ☐ affect the SER.

EVALUATED BY: [signature] (Name) Reactor Engineer (Title) 5/23/94 (Date)

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: [signature] RORY L. WARREN — NAME AND SIGNATURE — 5/23/94 DATE

APPROVED BY: [signature] — NAME AND SIGNATURE — 5/23/94 DATE

VI. Approved ☐ Rejected ☐
Reason for Rejection: ______

Licensing Engineer: ______ Date: ______ R3

1089 S00 PKG

are 125%. This ANSI testing is required to be performed pr[illegible] initial use, and following extensive repairs or modifications of the four Unit 1 handling devices utilized to handle fuel (S[illegible] Fuel Pit Bridge Crane, Refueling Machine, and 125T Auxiliary Bu[illegible] Crane main hook), as well as both hooks of the Polar Crane, have [illegible] successfully load tested at 125% rated capacity in prior tests. Additionally, this equipment has had no extensive maintenance or modifications which would affect structural integrity. Therefore, repeated load testing of this equipment is not warranted. The fourth device, the auxiliary hook of the 125T Auxiliary Building Crane, handles new fuel but has not been load tested at 125% capacity. Therefore it will be load tested to 125% of its ten-ton capacity.

(i) Appendix A, subparagraph 1.n.18

Because of its non-safety related, and simple functions, trace heating systems will receive component level tests, not preoperational or acceptance testing.

(j) Appendix A, subparagraph 2.b

Cold no flow, cold full flow and hot no flow rod drops do not provide any additional useful data. The drop times for these flow conditions are less conservative than for hot full flow conditions. We do not intend to perform cold no flow, cold full flow and hot no flow rod drops.

(k) Appendix A, subparagraphs 2.f and 5.m

WBN does not intend to perform a differential pressure measurement across the core nor a core flow measurement since these are prototype tests. WBN is not a prototype model plant, but, instead, is a well documented production model with several similar predecessor reactor units (Indian Point Unit 2, Trojan, and Sequoyah Unit 1) of its type. Proper differential pressures across the core and core flow is shown indirectly through performance of other tests that verify operating temperature and RCS flow.

(l) Appendix A, subparagraphs 4.c and 5.e

No new design information is to be gained from the RCCA Pseudo Ejection Test. WBN is not a prototype plant for 4 loop, 12 foot core with 17 x 17 fuel design and B_4C control rods. Performance and measurement data already exists for this design. Therefore, WBN will not perform this test.

(m) Appendix A, subparagraphs 4.a g. and 5.z ← Rory, This should be 4.g. Don

The proper response of process and effluent radiation monitors is demonstrated under the plant calibration program and during preoperational testing for the Process and Effluent Radiation Monitoring System (Sheet 31, Table 14.2-1). It is not expected that enough leakage in fuel cladding, steam generator tubes or heat exchangers will exist to provide a meaningful comparison between radiation monitor responses and laboratory analysis of samples.

TABLE 14.2-2

(Sheet 10 of 37)

ROD CONTROL SYSTEM
TEST SUMMARY

OBJECTIVE

To demonstrate that the rod control system satisfactorily performs the required control and indication functions.

PREREQUISITES

1. Rod position indication system testing has been completed.

2. The rod drop time measurements have been completed.

3. The reactor is in hot standby conditions at nominal operating temperature and pressure.

4. ~~The RCS boron concentration is equal to or~~ greater than the refueling boron concentration.

TEST METHOD

The test will be performed in hot standby and ensures that control room indicators respond properly and the control bank overlap function performs adequately. Rod bank starting and stopping positions will be compared with the control settings for verification. ~~The test will also verify the proper response of the control rod insertion alarms.~~ Each bank of shutdown rods will be operated individually in the withdraw and insert directions using the normal controls. The control banks will be operated in manual to verify the overlap function with minimum overlap settings. Sufficient travel will demonstrate drive operability, position indication, and other instrumentation without unduly increasing the count rate on any source channel above the established baseline rate.

ACCEPTANCE CRITERIA

None.

Note: The rod control system will be functionally tested to demonstrate:

1. The bank overlap functions.

2. Overall performance of the rod control system.

TABLE 14.2-2

(Sheet 17 of 37)

OPERATIONAL ALIGNMENT OF NUCLEAR INSTRUMENTATION
TEST SUMMARY

OBJECTIVE

To make adjustments to the source range, the intermediate range and power range channels and determine overlap between source/intermediate range and intermediate/power range channels.

PREREQUISITES

1. The nuclear instrumentation system source range, intermediate range and power range channels are installed and operable.

2. The reactor is stable at the test plateau.

3. Initial calibration has been implemented in accordance with Technical Specifications.

TEST METHOD

Initial settings for the source range channels are determined prior to core loading. Initial trip setpoints for the intermediate and power range channels are determined prior to power ascension and then adjusted to support power ascension. The overlap between source/intermediate range and intermediate/power range channels is determined during power ascension. Reactor power determined by secondary calorimetric is used to recalibrate (as needed) the intermediate and power range channels at each major test plateau. The power range channels are verified to be linear to ~~incore~~ power. ~~distribution, and channel outputs are calibrated at the 75% plateau.~~

ACCEPTANCE CRITERIA

1. All channels of source range, intermediate range and power range are operable and calibrated per Technical Specifications and Section 14.2.10.

2. Overlap between source range/intermediate range and between intermediate range/power range channels has been verified.

TABLE 14.2-2

(Sheet 20 of 37)

STARTUP ADJUSTMENTS OF REACTOR CONTROLS
TEST SUMMARY

OBJECTIVE

To determine the T_{avg} program resulting in the highest possible steam pressure and thus the optimum unit efficiency without exceeding pressure limitations for the turbine or the maximum allowable T_{avg}.

PREREQUISITES

1. The reactor is at hot standby conditions prior to initial criticality.

2. Recalibration of individual RTD resistance and temperature corrections from the performance of RTD-T/C Cross Calibration has been performed, if necessary.

3. The reactor/turbine control systems have been aligned to design values for turbine pressure and design value for T_{avg}.

TEST METHOD

The test will obtain primary system temperatures, steam pressures and thermal power data at steady state conditions. Evaluation of this data will provide the basis for the necessary instrument adjustments. ~~Zero power temperature and pressure data are recorded at hot standby conditions prior to criticality.~~ ~~Additional~~ Temperature and pressure data ~~are~~ is recorded at 30% and 50% power levels. Plots of RCS temperature versus percent power, steam generator pressure versus percent power, and turbine impulse chamber pressure versus percent power are then prepared. ~~After extrapolation to 100% power and analysis, the reactor and turbine control settings are revised to fit the new data, if necessary.~~ Additional data is taken at 75% power ~~and the process is repeated~~. This process is again ~~repeated~~ repeated at 100% power. ~~After all adjustments are made, a complete set of data is taken. This~~ to establish~~es~~ the optimum T_{avg} program for the reactor control system. H2f

ACCEPTANCE CRITERIA

1. Full load T_{avg} is less than or equal to design T_{avg}.

   NOTE: The program T_{avg} will be adjusted to the optimum steam generator load characteristic, such that steam pressure varies approximately from 1107 psia to 1000 psia for 0% to 100% power.

and then extrapolated to 100% power to determine if adjustments to the Tref program and/or calibration of the turbine impulse pressure may be required at 100% power.

TABLE 14.2-2 1089_S00 PKG

(Sheet 33 of 37)

TURNINE GENERATOR TRIP WITH COINCIDENT LOSS OF OFFSITE POWER TEST SUMMARY

OBJECTIVE

To demonstrate that the unit's response to a turbine generator trip with a coincident loss of offsite power is in accordance with design. This test will be performed during the 50% power testing plateau and be initiated from approximately 30% power.

PREREQUISITES

1. Reactor power is approximately 30% of rated thermal power.

2. The generator is synchronized to the TVA grid with a load of greater than or equal to 120 MWe.

3. The unit's electrical distribution is in its normal operating lineup. The RCP and unit station service transformers are carrying the ~~non-Class 1E~~ busses and the automatic transfers to offsite alternate power supplies are defeated.

4. The emergency diesel generators are in normal standby.

TEST METHOD

Initiate a manual turbine trip followed immediately with opening the breakers supplying offsite power to the common and shutdown ~~Class 1E~~ busses. Verify the emergency diesel generators start and load in the proper sequence. Demonstrate that the reactor coolant system can be maintained in a stable hot standby (Mode 3) condition for a minimum of 30 minutes using only the equipment available during a loss of offsite power.

ACCEPTANCE CRITERIA

1. The emergency diesel generators automatically start and load per design and provide power to the controls, indicators and equipment necessary to maintain the reactor coolant system in a safe hot standby condition for the duration of the test.

the common station service transformers are carrying the common and shutdown busses.

APPENDIX C

Page 1 of 1

1089 S01 PKG

FSAR CHANGE REQUEST

FSAR CHANGE PACKAGE NO. 1 0 8 9 1
Supp. No. R3

I. Attached are proposed changes to FSAR pages with justification for each change:

SECTION, PAGE NO. FIGURE OR TABLE	REASON FOR CHANGE
1) 14.2-31	1) & 2) The test method to Achieve Criticality is conservatively changed to Achieve Criticality At A higher rod configuration. This method provides the desired rod configuration Needed for the prerequisite to the next testing evolutions.
2) Table 14.2-2 Sheet 22 of 37	
3)	
4)	
5)	
6)	
7)	
8)	
9)	
10)	

(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
Technical Support	RORY L. WARREN	Rory L. Warren	5/27/94
Operations	D.M. Baker	D.M. Baker	6/1/94
Site Engineering	R.M. Johnson	R.M. Johnson	6/2/94
Site Licensing	C.L. Touchstone	C.L. Touchstone	6/1/94

CLT for JS 6/2/94

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☐ or DO NOT ☒ affect the SER.

EVALUATED BY: Whiting S. Delk (Name) RXE Supervisor (Title) 6-3-94 (Date)

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

None (Changes do not deviate from T/S's, or RG 1.68 R2) CLT 6-3-94

V. PREPARED BY: Thomas F. Huth Thomas F. Huth (NAME AND SIGNATURE) MAY 26, 1994 (DATE)

APPROVED BY: Whiting S. Delk Whiting S. Delk (NAME AND SIGNATURE) June 3 1994 (DATE)

VI. Approved ☐ Rejected ☐
Reason for Rejection: R3

Licensing Engineer: Date:

is actuated, core loading personnel will be evacuated. The situation will be evaluated before core loading is continued. After it has been determined that no hazards to personnel exist, personnel will be permitted to reenter the containment.

14.2.10.2 Postloading Tests

Upon completion of core loading, the reactor upper internals and the pressure vessel head will be installed. A test is conducted after filling and venting are completed to check the integrity of the vessel head installation.

Mechanical and electrical tests will be performed on the control rod drive mechanisms. These tests will include a complete operational checkout of the mechanisms and calibration of the individual rod position indication.

Tests will be performed on the reactor trip circuits to test manual trip operation. The actual control rod assembly drop times will be measured for each control rod assembly.

At all times that the control rod drive mechanisms are being tested, the boron concentration in the moderator will be maintained such that the shutdown margin requirements specified in the Technical Specifications are met. During individual RCCA or RCC bank motion, source range instrumentation is monitored for unexpected changes in core reactivity.

A functional electrical and mechanical check will be made of the incore nuclear flux mapping system. After evaluation of precritical tests, nuclear operation of the reactor will begin.

14.2.10.3 Initial Criticality

Initial criticality will be achieved by a combination of shutdown and control bank withdrawal and reactor coolant system boron concentration reduction.

Initially, all shutdown and ~~the shutdown banks will be withdrawn and the reactor coolant boron concentration adjusted before the approach to criticality begins, then~~ control banks will be withdrawn incrementally in the normal withdrawal sequence, leaving the last withdrawn control bank inserted far enough in the core to provide effective control when criticality is achieved. The boron concentration in the reactor coolant system will then be ~~further~~ reduced by the addition of primary water.

Inverse count rate ratio monitoring, using data from the source range instrumentation, will be used as an indication of the proximity and rate of approach to criticality. Inverse count rate ratio data will be plotted during control bank withdrawal and subsequent reactor coolant system boron concentration reduction.

Criticality will be achieved during boron dilution, or by ~~subsequent~~ rod withdrawal following boron dilution. The rate of primary water addition, and hence, the rate of approach to criticality will ~~may~~ be reduced as the reactor approaches criticality to ensure that effective control is maintained. Throughout this period, samples of the primary coolant are obtained and analyzed for boron concentration.

INITIAL CRITICALITY
TEST SUMMARY

OBJECTIVE

To achieve initial criticality in a cautious and controlled manner.

PREREQUISITES

1. Boron concentration in the RCS is conservative enough to preclude inadvertent criticality if the shutdown banks are at the all rods out position.

2. Source, intermediate and power range channels are calibrated and in operation. Power Range High Flux Trip setpoints are set to less than or equal to 20% power.

3. A minimum count rate of ½ cps and a signal to noise ratio of greater than 2 is verified on the source range channels.

4. The RCS is at hot no load pressure and temperature with a steam bubble in the pressurizer.

TEST METHOD

After establishing baseline count rates, the shutdown (if not previously pulled) and control banks are withdrawn in normal sequence until approximately 100 pcm worth remains inserted. RCS boron dilution is commenced at a rate of approximately 1000 pcm per hour ~~or less~~ with the RCS boron concentration being sampled at approximately 30 minute intervals. When the Inverse Count Rate Ratio (ICRR) is approximately 0.3, dilution rate is diminished by at least half of the original rate and ICRRs are renormalized to 1. Dilution is continued until the ICRR is approximately 0.2. Then dilution is terminated and the RCS is allowed to mix. If criticality is not achieved during mixing, the withdrawal of control bank D will be performed to achieve criticality. If criticality is not achieved when control bank D is fully withdrawn, control bank D will be reinserted to its original position. ICRR data will be renormalized and dilution will ~~commence at half of the last dilution rate. When ICRR reaches 0.2,~~ be initiated to a 20 to 5 pcm positive reactiv the dilution will then be ~~is~~ terminated. ~~and~~ The RCS is allowed to mix ~~to take the reactor critical. If criticality is not achieved,~~ then control bank D is withdrawn until criticality is achieved.*

Change made discussion w/ (Ray

100

During dilution, plots of ICRR versus time and ICRR versus dilution water additions are maintained.

ACCEPTANCE CRITERIA

1. The reactor achieves initial criticality in a safe and orderly manner.

* If criticality is not achieved when control bank D is fully withdrawn, control bank D will again be reinserted to its original position and the same small dilution followed by control bank D withdrawal evolution will be repeated until criticality is achieved.

APPENDIX C
Page 1 of 1

1089 S02 PKG

FSAR CHANGE REQUEST

FSAR CHANGE PACKAGE NO. 1 0 8 9   2 (Supp. No.)

R3

I. Attached are proposed changes to FSAR pages with justification for each change:

SECTION, PAGE NO. FIGURE OR TABLE	REASON FOR CHANGE
1) TABLE 14-2	Sheet 1 & 13 Title changed to clarify test.
2) Sheets 1 & 13	Sheet 13 Acceptance Criteria changed to 'NONE' And A Note Added clarify that the optimum settings are the goal of the test.
3) Sheet 10	Sheet 10 & 15 Although the rod drop timing may be performed 1st it is Not A true prerequisite to these tests.
4) Sheet 15	Sheet 16 The Test Method is reworded for clarity.
5) Sheet 16	Fig sheet 2 changed to match Test JFH 6/3/94
6) Figure 14.2-2 JFH 6/3/94	
7) Sheet 2	
8)	
9)	
10)	

(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
Technical Support (RBR 5/31/94, 5/31/94, 5/31/94)	Landy L McCormick	[signature]	5/31/94
Operations	S. M. Baker	[signature]	6/1/94
~~Division~~ Site Engineering	R. M. Johnson	[signature]	6/2/94
Site ~~Engineering~~ Licensing	C.W. Touchstone	[signature]	6/3/94

org changed JFH 6/3/94

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☐ or DO NOT ☒ affect the SER.

EVALUATED BY: [signature] WHITING S. DELK (Name) — RXE Supervisor (Title) — 6-3-94 (Date)

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

NONE (Changes do not deviate from T/S's, or RG 1.68 R2) [initials] 6-3-94

V. PREPARED BY: Thomas F. Huth [signature] (NAME AND SIGNATURE) — MAY 31, 1994 (DATE)

APPROVED BY: Whiting S. Delk [signature] (NAME AND SIGNATURE) — June 3, 1994 (DATE)

VI. Approved ☐ Rejected ☐
Reason for Rejection: ______

Licensing Engineer: ______ Date: ______

R3

1089 S02 PKG

TABLE 14.2-2

POWER ASCENSION TEST SUMMARIES

INDEX

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TABLE 14.2-2

ROD CONTROL SYSTEM
TEST SUMMARY

OBJECTIVE

To demonstrate that the rod control system satisfactorily performs the required control and indication functions.

PREREQUISITES

1. Rod position indication system testing has been completed.

~~2. The rod drop time measurements have been completed.~~

2 ~~3~~. The reactor is in hot standby conditions at nominal operating temperature and pressure.

3 ~~4~~. The RCS boron concentration is equal to or greater than the refueling boron concentration.

TEST METHOD

The test will be performed in hot standby and ensures that control room indicators respond properly and the control bank overlap function performs adequately. Rod bank starting and stopping positions will be compared with the control settings for verification. The test will also verify the proper response of the control rod insertion alarms. Each bank of shutdown rods will be operated individually in the withdraw and insert directions using the normal controls. The control banks will be operated in manual to verify the overlap function with minimum overlap settings. Sufficient travel will demonstrate drive operability, position indication, and other instrumentation without unduly increasing the count rate on any source channel above the established baseline rate.

ACCEPTANCE CRITERIA

None.

Note: The rod control system will be functionally tested to demonstrate:

1. The bank overlap functions.

2. Overall performance of the rod control system.

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TABLE 14.2-2

Capability

PRESSURIZER SPRAY AND CONTINUOUS
SPRAY FLOW SETTING
TEST SUMMARY

OBJECTIVE

To verify the effectiveness of the pressurizer spray. The throttle positions for the manual spray bypass valves are established and the setpoint for the Spray Line Low Temperature Alarm is determined.

PREREQUISITES

1. The reactor is in Mode 3 at normal operating temperature and pressure.

2. Pressurizer pressure and level instrumentation and associated control systems are calibrated.

TEST METHOD

~~Pressurizer spray effectiveness consists of a~~ transient initiated by full spray to reduce pressurizer pressure approximately 250 psi. Data is recorded during this pressure transient. The manual spray bypass valves are adjusted to an optimum position.

The setpoint for the Spray Line Low Temperature Alarm is set between 20°F and 50°F below the equilibrium temperature of the spray lines.

ACCEPTANCE CRITERIA

~~1. Pressurizer pressure response to the opening of both spray valves is within the allowable range specified by the NSSS vendor.~~

NONE

INSERT

NOTE: Pressurizer Spray does not provide a safety function nor is it required for the safe shutdown of the plant. Normal spray operation allows the plant to accept transients, which if occurred without normal spray, could trip the plant. As a result, Pressurizer Spray Capability will be evaluated against nominal NSSS performance curves. Optimum position for the manual spray bypass valves is achieved when during steady-state the Pressurizer Backup Heaters are not energized, and the spray line temperature is above the setpoint for the Spray Line Low Temperature Alarm. Deviations will be corrected per site procedures.

REACTOR COOLANT FLOW COASTDOWN
TEST SUMMARY

OBJECTIVE

To measure the rate at which reactor coolant flow changes subsequent to a simultaneous trip of all four reactor coolant pumps, to measure the delay times associated with assumptions of the loss of flow accident and measure the decay of the RCP voltage.

PREREQUISITES

1. RCS flow measurement testing has ~~and control rod drop time measurement testing have~~ been completed.

2. The reactor is at hot standby conditions with all rods inserted.

3. All RCPs are running and any pressure damping devices installed in the elbow tap differential pressure cell sensing lines for RCS flow measurement testing have been removed.

TEST METHOD

Measurements are made by tripping all reactor coolant pumps simultaneously and recording reactor coolant loop d/p, RCS low flow bistable position, reactor coolant pump breaker and reactor pump motor voltage decay data.

ACCEPTANCE CRITERIA

1. All reactor coolant pumps have been tripped within 100 msec of each other.

2. The reactor coolant flow coastdown is within the design of the coastdown flow values in Chapter 15.

3. The delay times associated with the low reactor coolant flow reactor trip are within the values assumed in Chapter 15.

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TABLE 14.2-2

(Sheet 16 of 37)

OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE INSTRUMENTATION
TEST SUMMARY

OBJECTIVE

To align the RCS ΔT and T_{avg} process instrumentation during power ascension.

PREREQUISITES

1. The reactor is stable at the test plateau.

2. The final calibration constants related to the primary loop RTDs (obtained from the RTD cross calibration test) have been entered into the process protection system for each RTD.

TEST METHOD

INSERT

~~Prior to initial criticality and with the RCS at isothermal conditions, RTD and process instrumentation temperature data is collected; and the ΔT and T_{avg} process instrumentation is aligned. At ascending power levels of 30%, 50% and 75%, similar temperature data is collected. At 75% plateau, the full power ΔT and T_{avg} values are determined by extrapolating the collected data to 100% power. The temperature instrumentation is then aligned for these values. At the 100% plateau, an alignment check of the ΔT and T_{avg} process instrumentation is performed. If required, new extrapolated values are determined. With satisfactory alignment results at 100% power, the reference T_{avg} parameter values may be used to rescale the Overpower ΔT trip setpoint.~~

ACCEPTANCE CRITERIA

1. Prior to criticality, RCS temperature channels ~~(T_{hot}, T_{cold}, T_{avg} and ΔT)~~ are aligned to within acceptable limits.

2. At 100% power, the process instrumentation value of ΔT for each channel is within 1% of reactor power as determined by a secondary calorimetric.

INSERT

TEST METHOD

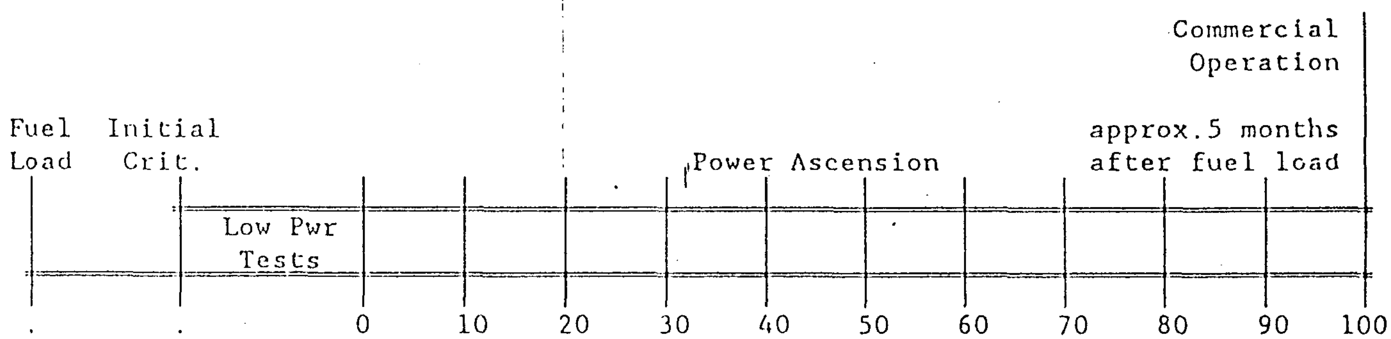
Prior to initial criticality with the RCS at isothermal conditions, temperature data is collected and the ΔT process instrumentation alignment is verified. At ascending power levels of 30%, 50%, and 75%, temperature, pressure, and calorimetric data is collected. This data is used to determine RCS hot and cold leg enthalpies. A curve fit, using the enthalpies and associated calorimetric power, is performed to predict the enthalpies for full power. This full power information is used to predict RCS temperatures (T_{hot}, T_{cold}, T_{avg}, and ΔT) at full power. Temperature instrumentation is aligned to these predicted values at 75% power. Then at 90% and 100% power, alignment checks of the ΔT and T_{avg} instrumentation are performed. If required, new values are calculated. With satisfactory alignment results at 100% power, the reference T_{avg} parameter values may be used to rescale the ΔT trip setpoints.

WBNP-84

FIGURE 14.2-2

(Sheet 2 of 2)

<u>POWER ASCENSION TEST SCHEDULE</u> (cont'd)




```
Shutdown From Outside the Control Rm -.--------.---------.----------------X (During the 100% plateau.)
T/G Trip With Coind. Loss Offsite Pwr-.--------.---------.----------------X (During the 50% plateau.)
Load Swing ---------------------------.--------.---------.--------------------------X------------------------X
Large Load Reduction -----------------.--------.---------.---------------------------------------------------X
Plant Trip From 100% Power (Ld Rej.) -.--------.---------.---------------------------------------------------X

RCS Flow Measurement -----------------.------X-.---------.--------------------------X------------X-----------X
Automatic Steam Gen. Level Control ---.--------.---------.--X-------------X---------X------------X-----------X
Dynamic Automatic Steam Dump Control -.--------.---------.---X
Automatic Reactor Control System -----.--------.---------.--------------------------X

Thermal Expan. of Piping Systems -----.--X-----.---------.----------------X---------X------------X-----------X
Pipe Vibration Monitoring ------------.--X-----.---------.----------------X---------X------------X-----------X
Oper. Align. of Nuclear Inst. --------.------X-.---X-----.----------------X---------X------------X-----------X
Loose Parts Monitoring System --------.--------.---------X----------------X---------X------------X-----------X
Oper. Align. of Process Temp. Inst. --.------X-.---------.----------------X---------X------------X-----X-----X
Startup Adjust. of Reactor Control ---.------X-.---------.----------------X---------X------------X-----------X
Calib. of Steam and Feedwater Flow ---.------X-.---------.----------------X---------X------------X-----------X
```

Licensing Comment

"X" placed At 90% for Oper. Align of Process Temp.

APPENDIX C

Page 1 of 1

1089 S03 PKG

FSAR CHANGE REQUEST

FSAR CHANGE PACKAGE NO. 1 0 8 9 S 3
Supp. No.

R3

I. Attached are proposed changes to FSAR pages with justification for each change:

SECTION, PAGE NO. FIGURE OR TABLE	REASON FOR CHANGE
1) Table 14.2-2, Sheet 33	Clarify the Acceptance Criteria to emphasis the ability to control the unit with only onsite power sources rather than the loading of the diesel generators, which is tested elsewhere.
2) N/A RAW 6/10/94	N/A RAW 6-10-94
3)	
4)	
5)	
6)	
7)	
8)	
9)	
10)	

(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
ENGINEERING	FA Koontz Jr for RHJ...	[signature]	6-10-94
~~OPS~~	D.A. Kulisek	[signature]	6/10/94
LICENSING	Charlie Touchstone	CW Touchstone	6/10/94

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☐ or DO NOT ■ affect the SER.

EVALUATED BY: [signature] (Name) RXE Supervisor (Title) 6-10-94 (Date)

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

NONE

V. PREPARED BY: [signature] RORY L. WARREN (NAME AND SIGNATURE) 6-10-94 (DATE)

APPROVED BY: DENNIS L. KOEHL [signature] (NAME AND SIGNATURE) 6/10/94 (DATE)

VI. Approved ☐ Rejected ☐
Reason for Rejection: ________

Licensing Engineer: ________ Date: ________

R3

TABLE 14.2-2
(Sheet 33 of 37)

TURBINE GENERATOR TRIP WITH
COINCIDENT LOSS OF OFFSITE
POWER TEST SUMMARY

U/C RECOMMENDS STET per RG words 5.JJ

OBJECTIVE

To demonstrate the unit's response to ~~a~~ turbine generator trip with a coincident loss of offsite power ~~is in accordance with design~~. This test will be performed during the 50% power testing plateau and be initiated from approximately 30% power.

PREREQUISITES

1. Reactor power is approximately 30% of rated thermal power.

2. The generator is synchronized to the TVA grid with a load of greater than or equal to 120 MWe.

3. The unit's electrical distribution is in it's normal operating lineup. The unit station service transformers are carrying the RCP buses and unit buses~~busses~~ and the common station service transformers are carrying the common and shutdown buses~~busses~~. The automatic transfers to offsite alternate power supplies are defeated.

4. The emergency diesel generators are in normal standby.

TEST METHOD

Initiate a manual turbine trip followed immediately with opening the breakers supplying offsite power to the common and shutdown buses~~busses~~. The~~Verify the~~ emergency diesel generators will start and load ~~in the proper sequence~~. Maintain~~Demonstrate that~~ the Reactor Coolant System~~reactor coolant system can be maintained~~ in ~~a stable~~ hot standby (Mode 3) conditions for a minimum ~~condition for a minimum~~ of 30 minutes using only the equipment available during the~~a~~ loss of offsite power. Onsite power systems (emergency diesel generators and batteries) provide the necessary power to controls, indicators, and equipment for the duration of the test.

ACCEPTANCE CRITERIA

1. Safety injection is not initiated.

2. Pressurizer and steam generator safety valves do not open.

3. Monitored plant parameters can be maintained in hot standby conditions using only the equipment available during the loss of offsite power.

~~The emergency diesel generators automatically start and load per design and provide power to the controls, indicators and equipment necessary to maintain the reactor coolant system in a safe hot standby condition for the duration of the test.~~

APPENDIX C

1096 S00 PKG

Page 1 of 1

FSAR CHANGE REQUEST

FSAR CHANGE PACKAGE NO. 1 0 9 6 ______ Supp. No. R3

I. Attached are proposed changes to FSAR pages with justification for each change:

SECTION, PAGE NO. FIGURE OR TABLE	REASON FOR CHANGE
1) SEE ATTACHED	
2)	
3)	
4)	
5)	
6)	
7)	
8)	
9)	
10)	

(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
OPERATIONS	David S. King	[signature]	6/21/94
TECH SUPPORT CCP	Landy L. McCormick	[signature]	6/21/94
ENGINEERING			
LICENSING			

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☐ or DO NOT ☒ affect the SER.

EVALUATED BY: [signature] /Name SUT ENGR Title 6/14/94 /Date

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: [signature] NAME AND SIGNATURE 6/14/94 DATE

APPROVED BY: [signature] NAME AND SIGNATURE 6/15/94 DATE

VI. Approved ☐ Rejected ☐
Reason for Rejection: ____________

Licensing Engineer: ____________ Date: ____________ R3

APPENDIX C

Page 1 of 1

1096 S00 PKG

FSAR CHANGE REQUEST

FSAR CHANGE PACKAGE NO. 1096 ______ Supp. No. R3

I. Attached are proposed changes to FSAR pages with justification for each change:

SECTION, PAGE NO. FIGURE OR TABLE	REASON FOR CHANGE
1) SEE ATTACHED	
2)	
3)	
4)	
5)	
6)	
7)	
8)	
9)	
10)	

(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
OPERATIONS			
TECH SUPPORT			
ENGINEERING	FAKOONTZ JR RMJ	[signature]	6-28-94
LICENSING			

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☐ or DO NOT ☒ affect the SER.

EVALUATED BY: [signature] SUT ENGR 6/14/94
Name Title Date

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: [signature] 6/14/94
NAME AND SIGNATURE DATE

APPROVED BY: [signature] 6/15/94
NAME AND SIGNATURE DATE

VI. Approved ☐ Rejected ☐
Reason for Rejection: ______

Licensing Engineer: ______ Date: ______ R3

APPENDIX C

Page 1 of 1

1096. S00 PKG

FSAR CHANGE REQUEST

FSAR CHANGE PACKAGE NO. 1096 ______ Supp. No. R3

I. Attached are proposed changes to FSAR pages with justification for each change:

SECTION, PAGE NO. FIGURE OR TABLE	REASON FOR CHANGE
1) SEE ATTACHED	
2)	
3)	
4)	
5)	
6)	
7)	
8)	
9)	
10)	

(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
~~OPERATIONS~~			
~~TECH SUPPORT~~			
~~ENGINEERING~~			
~~LICENSING~~	T.L. Porter	TLPorter	7/6/94

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☐ or DO NOT ☒ affect the SER.

EVALUATED BY: [signature] /Name   SUT ENGR Title   6/14/94 /Date

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

V. PREPARED BY: [signature] NAME AND SIGNATURE   6/14/94 DATE

APPROVED BY: [signature] NAME AND SIGNATURE   6/15/94 DATE

VI. Approved ☐ Rejected ☐ R3
Reason for Rejection: ______

Licensing Engineer: ______ Date: ______

FSAR Chapter 14
Reason for Changes By Page
TABLE 14.2-1

Sheet	
40	To delete reference to previous data for comparison. PTI data will become the basis for comparison with future data. In addition to clarify that voltage, current and temperature are all involved in the Test Method and Acceptance Criteria, and that testing is done on a circuit (two igniters), rather than individual igniter basis. See draft DCN on this subject, dated 6/14/94 (attached).
48	See DCN Q-30919-A (attached)
85	To reflect the current system design basis. See note from Arnold to Ondriska dated 6/15/94 (attached).

COMBUSTIBLE GAS CONTROL SYSTEMS
TEST SUMMARY

OBJECTIVE

To demonstrate the proper operation of the Hydrogen Recombiners, and Hydrogen Mitigation System (HMS) including the heaters, controllers, igniters, fans, and instrumentation.

To demonstrate the ability of the hydrogen sampling system to detect the presence of hydrogen in the primary containment atmosphere and give indication in the control room.

PREREQUISITES

1. Each igniter has been energized for the minimum required duration and allowed to cool prior to test conduct.

2. Containment Ventilation Systems are operational.

TEST METHOD

1. Verify design air flow through hydrogen recombiners and sample tubing.

2. Verify the hydrogen recombiner heatup rate and operating outlet temperature.

3. Verify sample system controls, interlocks, instrumentation, and alarms operation.

4. Demonstrate the hydrogen analyzer is capable of drawing and analyzing a sample and returning it to containment. Verify alarms and instrumentation function properly.

5. Energize each HMS igniter CIRCUIT and verify the voltage, CURRENT, and IGNITER surface temperature.

ACCEPTANCE CRITERIA

1. The Hydrogen Recombiners will achieve and maintain operating temperature at design air flow at design power input as described in FSAR Section 6.2.5.

2. The hydrogen analyzer system is capable of sampling and analyzing the containment atmosphere in accordance with design requirements as described in FSAR Section 6.2.5.

3. All HMS igniters are operational AT A MINIMUM REQUIRED SURFACE TEMPERATURE WITHOUT EXCEEDING A MAXIMUM POWER REQUIREMENT CALCULATED FROM CIRCUIT ~~with~~ current and voltage readings, ~~comparable to previous test data in accordance with FSAR Section 6.2.5.~~ IN ACCORDANCE WITH THE APPLICABLE DESIGN DOCUMENTS.

4. Controls, interlocks, instrumentation, and alarms operate in accordance with FSAR Section 6.2.5 and design drawings.

TABLE 14.2-1

(Sheet 48 of 90)

AC POWER DISTRIBUTION SYSTEM
TEST SUMMARY

TEST METHOD (Cont'd)

5. Demonstrate manual and automatic transfer schemes operate in accordance with design drawings.

6. Verify each of the offsite power sources provide the required no-load voltage to assigned 6.9 kV boards and in turn to the 480V boards.

7. Verify proper load shedding and/or undervoltage protection of all 6.9 kV shutdown boards and 480 Volt shutdown boards and confirm diesel generator units receive proper start signals.

~~8. Verify the capability of each common station transformer to carry the load required to supply ESF loads of one unit under loss of coolant accident conditions in addition to power required for shutdown of the non-accident unit.~~

~~9.~~ 8. Select the Class 1E train having the lowest analyzed voltage and record grid, 6900 and 480 volt bus parameters at no-load, steady state (minimum 30% of worst case load), and transient conditions. Induce the transient by the non-concurrent start of a Class 1E 6.9kv motor and a non-Class 1E 6.9kV motor.

NOTE: Vital 120 volt AC power voltage surveys will be performed in conjunction with this test.

1096 S00 PKG

TABLE 14.2-1

ANTICIPATED TRANSIENT WITHOUT SCRAM MITIGATION SYSTEM
ACTUATION CIRCUITRY
TEST SUMMARY

OBJECTIVE

Demonstrate the capability of the Anticipated Transients Without Scram Mitigation System Actuation Circuitry (AMSAC) to respond properly to initiation signals.

PREREQUISITES

1. Interfacing systems such as the Annunciator System are available.

2. Instrument loops providing input to AMSAC.

TEST METHOD

1. Simulating transmitter input, verify AMSAC functions properly with the appropriate ~~turbine impulse pressure~~ (simulated power), steam generator level, and block switch position conditions.

2. Demonstrate AMSAC setpoints and time response requirements.

3. Verify proper annunciation from AMSAC System.

ACCEPTANCE CRITERIA

1. When AMSAC is armed and Steam Generator level coincidence logic is obtained, a Main Turbine trip signal and start signal for all Auxiliary Feedwater Pumps is generated. AMSAC test/block switch can block an AMSAC signal.

2. With AMSAC armed at ≥40% simulated power, Steam Generator low-low level logic setpoint is ≥12% of narrow range level. ~~and less than the RPS low-low level trip setpoint.~~ With AMSAC armed at ≥80% simulated power, Steam Generator low-low level logic setpoint is ≥25% of narrow gauge level and less than the RPS low-low level trip setpoint.

restatement of design

3. AMSAC logic and output relay actuation response time is ≤1.0 seconds. AMSAC actuation time delay including sensor, logic and output relay actuation response time is ≤30 seconds.

4. AMSAC status lights and annunciators respond as designed.

N ORIGINAL INITIATING DESIGN CHANGE NOTICES

APPENDIX A
Page 1 of 6

ISSUE RIMS ____ CLOSURE RIMS 94-0381

1 DCN TYPE M [ ] S [ ] F [ ] W [ ] Q [x]	DESIGN CHANGE NOTICE	2 DCN NO. 3 PAGE ~~1~~ 2 ccl 5/2/94

PART I REQUESTED CHANGE

4 PLANT/UNIT WATTS BAR NUCLEAR PLANT UNIT - 1	5 REASON CODE NONE
6 AREA/BLDG LOC Transformer Yard Area	7 EQUIP ID NO (S)/SYS CODE(S) COMMON STA SERV XFMR C/COMMON STA SERV XFMR D/SYS 200
8 AUTHORIZING DOCUMENT FSAR TABLE 14.2-1, Sheet 47, AC POWER DISTRIBUTION SYSTEM TEST SUMMARY (Test Method 8)	9 REFERENCE DOCUMENTS 1-15E500-2, 1-15E500-3, FSAR 8.1.4
10 DESCRIPTION SUMMARY ~~Test Method 8 for the~~ AC POWER DISTRIBUTION SYSTEM TEST SUMMARY requires testing to demonstrate the capability of each Common Station Service Transformer (CSST) to supply ESF (LOCA) loads on one unit and shutdown loads on the other unit.	11 REMARKS

12 DESCRIPTION OF PROBLEM/REQUESTED CHANGE

ADVANCED AUTHORIZATION REQUESTED YES [ ] NO [ ]

Startup requests from Engineering the actual load (in KW) required for each Train in both Unit 1 and Unit 2 to perform the required testing. Startup also requests engineering assistance in determining the appropriate acceptance criteria values to apply for this test scenario.

13 JUSTIFICATION/REASON FOR CHANGE
To ensure the proper load values and acceptance criteria parameters are utilized.

14 REQUESTED BY	ORGANIZATION	EXT	DATE	NEED DATE	14A ORIGINATOR'S SUPV	DATE
G. Grizard/ J. Dorn	Startup	1461 /1073	4/25/94	5/6/94	G. Ondriska	4/26/94

15 REVIEWED BY RLE	DISCIPLINE ASSIGNMENT	DATE	15A PROJECT MANAGER APPROVAL / PNL No.	DATE
			15B ENGINEERING APPROVAL TO INITIATE	DATE

PART II APPROVED CHANGE

16 PROBLEM SOLUTION/APPROVED CHANGE (INCLUDE BASIS FOR APPROVAL)		ADVANCE AUTHORIZATION APPROVED YES [ ] NO [ ] AUTHORIZING ENGINEER	DATE
17 TE	DATE	22 DESIGN VERIFIER	DATE
18 OTHER	DATE	23 RLE	DATE
19 OTHER	DATE	24 EM	DATE
20 OTHER	DATE	25 WORK COMPLETION	DATE
21 QA	DATE	26 FINAL WORK TRACK. CLOSURE	DATE

ccl 5/2/94

94-0384

Startup requests from engineering the actual load (in KW) required for each train in both Unit 1 and Unit 2 to perform the required testing. Startup also requests engineering assistance in determining the appropriate acceptance criteria values to apply for this test scenario.

Each of the 6.9KV secondary windings on Common Station Service Transformers (CSST) C & D have OA/FOA/FOA ratings of 24/32/40 MVA. The CSSTs corresponding primary winding ratings are 33/44/55 MVA. All four 6.9 KV Shutdown Boards can be connected to the same CSST (either C or D) with all accident loads operating in both units (not a design basis), and the CSST's OA ratings are not challenged. ~~Therefore,~~ CSST C and D power ratings are not a concern, and there is no need to test the transformers' power rating.

The voltage relays associated with the load tap changers (LTC) on the secondary side of CSST C and D monitor the voltage on the 6.9KV Shutdown Board and maintain the board voltage via the LTC within the nominal limits specified on drawing 1-15E500-3. Since the voltage relays monitor the voltage at the shutdown boards, CSST transformer and cable losses (from the CSST to the shutdown board) are compensated for and pre-accident loading on CSST C and D has little or no effect on board voltage for transient/ steady-state accident conditions.

1096 S00 PKG

FOR	NAME: George Ondriska	DATE: 6/15/94
	ADDRESS: Watts Bar Startup	☐ Chatta ☐ M. S. ☐ Knox ☐ Nor.

Fold here for return

FROM	NAME: Larry Arnold	EXTENSION:
	ADDRESS: Watts Bar Startup	☐ Chatta ☐ M. S. ☐ Knox ☐ Nor.

George - Please see the attached FSAR Test Summary (Sheet 85 of 90) for AMSAC. The portion of Acceptance Criteria 2 concerning the Steam Generator Level Setpoint at ≥ 80% power is no longer applicable. This breakpoint has been deleted from the AMSAC system. The Design Criteria Document and the AMSAC Test Summary were updated in May 1995. Section 7.7.1.12 of FSAR which describes AMSAC does not mention the ≥80% power setpoint and is satisfactory as written.

APPENDIX A 1096 S00 PKG
Page 1 of 6

ISSUE
RIMS________________

CLOSURE
RIMS________________

1 DCN TYPE M [ ] S [ ] F [ ] W [ ] Q [ ]	DESIGN CHANGE NOTICE	2 DCN NO. 3 PAGE 1

PART I REQUESTED CHANGE

4 PLANT/UNIT WATTS BAR NUCLEAR PLANT UNIT - 1	5 REASON CODE NONE
6 AREA/BLDG LOC Containment Hydrogen Igniters	7 EQUIP ID NO (S)/SYS CODE(S) Hydrogen Mitigation/System 268
8 AUTHORIZING DOCUMENT PTI-268-01	9 REFERENCE DOCUMENTS System Description N3-83-4001
10 DESCRIPTION SUMMARY Revise N3-83-4001, System Description for Combustible Gas Control	11 REMARKS
12 DESCRIPTION OF PROBLEM/REQUESTED CHANGE	ADVANCED AUTHORIZATION REQUESTED YES [ ] NO [ ]

Paragraph 2.2.14.5 of System Description N3-83-4001 indicates that the maximum power requirement of each igniter shall not exceed 600 watts. FSAR 6.2.5A.5 states that testing for Hydrogen Mitigation System (HMS) consists of energizing the system and taking voltage and current readings from each circuit at the distribution panels. The specific power requirement of an individual igniter can not be determined by measurements at the distribution panel since there are two igniters per circuit. Revise N3-83-4001 to indicate 1200 watts per circuit (average 600 watts per igniter). Also, FSAR Table 14.2-1, Sheet 40, Prerequisite #1, indicates a minimum required energization duration and then a cooling period prior to testing the igniters. Provide a design output to specify the minimum energized and minimum cooling durations.

13 JUSTIFICATION/REASON FOR CHANGE

Clarification to System Description is required to resolve Peer Review comments for Preoperational Test Procedure PTI-268-01.

14 REQUESTED BY	ORGANIZATION	EXT	DATE	NEED DATE	14A ORIGINATOR'S SUPV	DATE
Larry R. Coleman	SUT	2077	6/14/94			

15 REVIEWED BY RLE	DISCIPLINE ASSIGNMENT	DATE	15A PROJECT MANAGER APPROVAL	PWL No.	DATE
			15B ENGINEERING APPROVAL TO INITIATE		DATE

PART II APPROVED CHANGE

16 PROBLEM SOLUTION/APPROVED CHANGE (INCLUDE BASIS FOR APPROVAL)	ADVANCE AUTHORIZATION APPROVED: YES [ ] NO [ ] AUTHORIZING ENGINEER — DATE

17 TE	DATE	22 DESIGN VERIFIER	DATE
18 OTHER	DATE	23 RLE	DATE
19 OTHER	DATE	24 EM	DATE
20 OTHER	DATE	25 WORK COMPLETION	DATE
21 QA	DATE	26 FINAL WORK TRACK. CLOSURE	DATE

1097 S00 PKG

JUN 23 1994   T33 940623 879

Bruce S. Schofield, FSB 2K, Watts Bar Nuclear Plant

WATTS BAR NUCLEAR PLANT (WBN) - FINAL SAFETY ANALYSIS REPORT (FSAR)
TRANSMITTAL PACKAGE 1097 SO

The purpose of this memorandum is to provide to you FSAR Transmittal Package 1097 SO

This package contains approved Nuclear Engineering (NE) design input in accordance with ANSI N45.2.11 for incorporation into the FSAR. Subsequently, this material should not be changed without approval. These changes have been reviewed by NE lead and support organizations and approval signatures are included. The changes do not have the potential to affect the conclusions stated in the Safety Evaluation Report (SER), and do not modify a commitment to the Nuclear Regulatory Commission (NRC).

R. M. Johnson
Acting Engineering Manager
IOB 1A-WBN

Attachments
cc (Attachments):
R. B. Yager, IOB 1H-WBN
RIMS, QAC 1G-WBN

3127S

APPENDIX C

Page 1 of 1

1097 S00 PKG

FSAR CHANGE REQUEST

FSAR CHANGE PACKAGE NO. 1 0 9 7 S0
Supp. No.

R3

I. Attached are proposed changes to FSAR pages with justification for each change:

SECTION, PAGE NO. FIGURE OR TABLE	REASON FOR CHANGE
1) Sect 6.2.6.1, p. 6.2.6-1 & -2	This change is required to clarify the containment leak rate methodology so as to accurately represent the test criteria & to meet the design requirements.
2) Table 14.2-1 Sheet 82	
3)	
4)	
5)	
6) Table 6.2.6-3 Sheets 1 thru 8	Corrected Lineup Statuses for X-27A,B,C,D, X-84B,C,D, X-87B,C,D and X-58B. Added Notes and other minor clarifications to better define status of lineups.
7)	
8)	
9)	
10)	

(Provide attachments if additional listings are required)

II. The proposed changes have been coordinated with and are concurred by the supporting organizations

SUPPORTING ORG	NAME	Signature	Date
See Attachment 5			
	JSJ 6-17-94		

III. The proposed changes have been evaluated with respect to potential effects on conclusions drawn in the Safety Evaluation Report (SER). Changes DO ☐ or DO NOT ☐ affect the SER.

EVALUATED BY: Jeffrey S Thompson (Name) Mech Engr - NE-LMN (Title) 6/17/94 (Date)

IV. References (DCNs, SCRs, SCARs, PERs, etc.)

None

JSJ 6-17-94

V. PREPARED BY: See Attachment 5
NAME AND SIGNATURE — DATE

APPROVED BY: JSJ 6-17-94
NAME AND SIGNATURE — DATE

VI. Approved ☐ Rejected ☐
Reason for Rejection:

Licensing Engineer: Date:

R3

CHANGE PACKAGE NO. 1097 SO

NUCLEAR ENGINEERING 1097 S00 PKG
FSAR CHANGE REQUEST

ADDITIONAL INFORMATION FORM

PART I - NE LEAD ORGANIZATION REVIEW

A. Does this change *modify* a previous NRC commitment?

( ) YES (X) NO

If YES: Ensure compliance with the requirements of EAI-3.08 and SSP-4.03

B. Is there a DCA that affects an FSAR figure? (i.e. there is an FSAR figure based on a TVA design drawing, and that drawing is to be revised per the DCA)

( ) YES (X) NO

If YES: FSAR figure no. ______________ TVA drawing no. ______________

SEE NEXT SHEET FOR PARTS II, III, AND IV

1097 S00 PKG CHANGE PACKAGE NO. 1097 SO

NUCLEAR ENGINEERING FSAR CHANGE REQUEST ADDITIONAL INFORMATION FORM

PART II - NE LEAD ORGANIZATION SIGNATURES

NOTE: The LO "reviewed" signature signifies that a design verification review was performed per EAI-3.06 guidelines.

	NE Section	Name	Signature	Date	Phone
INITIATED BY:	LMN-NSSS	Jeffery S. Thompson	J S Thompson	6-22-94	X1583
REVIEWED BY:	LMN-NSSS	J. Steve Robertson	J. Steve Robertson	6-22-94	X1601
APPROVED BY:	LMN-NSSS	R.E. Wiggall	R E Wiggall	6/23/94	X1593

PART III - NE SUPPORT REVIEW - REQUIRED FOR ALL CHANGES**

**(REFER TO ATACHMENT 13 [DOR] FOR APPLICABILITY)

NOTE: NE support signatures signify that an interface review was performed per EAI-3.06 guidelines.

	NE Section	Name	Signature	Date	Phone
NE - CE:		Not Req'd JST 6-22-94			
NE - EE:					
NE - MN:	NSSS	Not Required - initiating org	JST	6-22-94	
NE - OE:	SA	Not Req'd JK 6-23-94			
NE - OE:	MAT				
NE -		Not Req'd JST 6-22-94			
NE -					

PART IV - EXTERNAL SUPPORT REVIEW - REQUIRED FOR ALL CHANGES**

**(REFER TO ATTACHMENT 13 [DOR] FOR APPLICABILITY)

( ) *No external review required per DOR.*

Organization	Name	Signature	Date	Phone
SYSENG, TSS, ESS, RXE	Dennis L Krehl	Dennis Krehl	6/23/94	6775
Ops	See Attached			
Startup & Test	M.S. Batestani	M S Batestani	6/23/94	5282

1097 S00 PKG CHANGE PACKAGE NO. 1097 S0

NUCLEAR ENGINEERING FSAR CHANGE REQUEST ADDITIONAL INFORMATION FORM

PART II - NE LEAD ORGANIZATION SIGNATURES

NOTE: The LO "reviewed" signature signifies that a design verification review was performed per EAI-3.06 guidelines.

	NE Section	Name	Signature	Date	Phone
INITIATED BY:					
REVIEWED BY:					
APPROVED BY:					

PART III - NE SUPPORT REVIEW - REQUIRED FOR ALL CHANGES**

**(REFER TO ATACHMENT 13 [DOR] FOR APPLICABILITY)

NOTE: NE support signatures signify that an interface review was performed per EAI-3.06 guidelines.

See Cooley 6-23-94

	NE Section	Name	Signature	Date	Phone
NE - CE:					
NE - EE:					
NE - MN:					
NE - OE:					
NE -					
NE -					
NE -					

PART IV - EXTERNAL SUPPORT REVIEW - REQUIRED FOR ALL CHANGES**

**(REFER TO ATTACHMENT 13 [DOR] FOR APPLICABILITY)

( ) *No external review required per DOR.*

Organization	Name	Signature	Date	Phone
LIC	T. L. Porter	[signature]	6/23/94	3854
RES Group	N/A - No impact on this org	[signature]	6-23-94	

1097 S00 PKG CHANGE PACKAGE NO. 1097 SO

NUCLEAR ENGINEERING FSAR CHANGE REQUEST ADDITIONAL INFORMATION FORM

PART II - NE LEAD ORGANIZATION SIGNATURES

NOTE: The LO "reviewed" signature signifies that a design verification review was performed per EAI-3.06 guidelines.

	NE Section	Name	Signature	Date	Phone
INITIATED BY:	LMN-NSSS	Jeffery S. Thompson	JS Thompson	6-22-94	X1583
REVIEWED BY:					
APPROVED BY:					

See Attached JST 6-23-94

PART III - NE SUPPORT REVIEW - REQUIRED FOR ALL CHANGES**
**(REFER TO ATACHMENT 13 [DOR] FOR APPLICABILITY)

NOTE: NE support signatures signify that an interface review was performed per EAI-3.06 guidelines.

	NE Section	Name	Signature	Date	Phone
NE - CE:		Not Req'd JST 6-22-94			
NE - EE:					
NE - MN:	NSSS	Not Required - initiating org	JSJ	6-22-94	
NE - OE:	SA				
NE - OE:	MAT				
NE -		Not Req'd JST 6-22-94			
NE -					

PART IV - EXTERNAL SUPPORT REVIEW - REQUIRED FOR ALL CHANGES**
**(REFER TO ATTACHMENT 13 [DOR] FOR APPLICABILITY)

( ) *No external review required per DOR.*

Organization	Name	Signature	Date	Phone
SYS ENG, TSS, ESS, RXE	See Attached			
Ops	R.G. MENDE	RG Mende	6-22-94	7733
Startup & Test	See Attached			

1097 S00 PKG

6.2.6 Containment Leakage Testing

Primary containment leakage tests and containment isolation system valve operability tests will be performed periodically to verify that leakage from the containment is maintained within acceptable limits set forth in the Technical Specifications. The types of leakage tests are as follows:

1. Test Type A

   Tests to measure the reactor primary containment overall integrated leakage rate. The containment leak rate test will be conducted in accordance with 10 CFR 50, Appendix J.

2. Test Type B

   Tests to detect and measure local leaks of containment penetrations, hatches, and personnel locks as required by 10 CFR 50, Appendix J.

3. Test Type C

   Test to detect and measure containment isolation valve leakage as described by 10 CFR 50, Appendix J.

The leakage rate testing pressure for the above tests, P_a (as defined in 10CFR50 Appendix J), has a nominal value of 15.0 psig with allowance for instrument error. Exceptions to this test pressure are noted elsewhere in this section.

6.2.6.1 Containment Integrated Leak Rate Test

REPLACE WITH INSERT A.

The maximum allowable containment leakage rate for the Watts Bar Nuclear Plant is 0.25 weight percent per day as specified in the Technical Specifications. The test will be conducted in full compliance with 10 CFR 50, Appendix J, with the following exception. The minimum test duration, after at least a four hour stabilization period, will be at least 8 hours in accordance with Bechtel Topical Report BN-TOP-1, Rev. 1, "Testing Criteria for Integrated Leakage Rate Testing of Primary Containment Structures for Nuclear Power Plants."

Prior to conducting the integrated leak rate test, those lines which penetrate primary containment are aligned as shown in Table 6.2.6-3.

The containment is then pressurized in accordance with 10 CFR 50, Appendix J, and the Technical Specification requirements. When test pressure is reached, the containment is isolated from its pressure source and the following parameters are recorded at periodic intervals:

1. Containment absolute pressure
2. Dry bulb temperatures
3. Water vapor pressures
4. Outside containment ~~weather~~ pressure and temperature conditions

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INSERT A

The maximum allowable containment leakage rate for the Watts Bar Nuclear Plant is 0.25 weight percent per day as specified in the Technical Specifications. The preoperational testing will be conducted in full compliance with 10CFR 50, Appendix J as shown in Table 14.2-1. ~~sheet 82.~~ Licensing editorial comment. Subsequent periodic testing will also be performed in accordance with Appendix J. Periodic testing durations of less than 24 hours may be conducted when performed in accordance with Bechtel Topical Report BN-TOP-1, Revision 1, "Testing Criteria for Integrated Leakage Rate Testing of Primary Containment Structures for Nuclear Power Plants."

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During the test, ventilation inside the containment is operated as necessary to enhance an even air temperature distribution. The test data are processed at periodic intervals during the test to determine test status and leakage conditions. If it appears that the leakage is excessive, the pressure plateau is either maintained on the test or aborted to perform repairs. The test is run for a prescribed time period to obtain assurance of the leak test rate.

Following the leak rate test, a second leak rate is performed to verify the information obtained in the first test. This verification test consists of slowly bleeding off pressure from containment at a known rate and measuring the total containment leak rate. The superimposed, measured flow is adjusted to a value which causes a change in the weight of air in the containment that is in the same order of magnitude as the allowable leakage rate.

The total time equations or the mass point equations are ~~point-to-point method is~~ used to determine the integrated leak rate. ~~This method is discussed in the Bechtel Topical Report BN-TOP-1, Rev. 1, "Testing Criteria for Integrated Leak Rate Testing of Primary Containment Structures for Nuclear Power Plants." In addition to this method, the mass point method is also used to determine the integrated leak rate. The formulas used for the mass point method are those formulas found in ANSI/ANS-56.8, "Containment System Leakage Testing Requirements."~~ The mass point equations will be used for preoperational testing as discussed in Table 14.2-1. ~~Sheet 82.~~

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6.2.6.2 Containment Penetration Leakage Rate Test

Table 6.2.4-1 lists penetrations in the primary containment. The Type B test is performed on all operational electrical equipment and personnel hatch, fuel transfer tube, thimble renewal, and ice blowing penetrations, and penetration bellows in accordance with 10 CFR 50, Appendix J. The dual-ply bellows on containment penetration will be tested at P_a by applying the pressure between the plies. Airlock door seals are tested at 6.0 psig per Technical Specification requirements. Experience has shown that pressurizing the space between the seals to greater than 6.5 psig on personnel airlock doors of the design used at Watts Bar will lift the door and induce gross leakage unless strongbacks are used. Since the door seal test is intended to prove integrity of the seals, it is our position that a test conducted at 6.0 psig will conservatively demonstrate that seal integrity is maintained.

Table 6.2.6-1 lists all penetrations subjected to type B testing. Spare electrical penetrations will be subjected to Type B testing as they become operational. Tables 6.2.4-1 through 6.2.4-4 and Figures 8.3-44 and 8.3-45 give details on these penetrations. The test is performed in full compliance with 10 CFR 50, Appendix J. The acceptance criteria as required by Appendix J are specified in the Technical Specifications.

Table 6.2.4-1 lists containment isolation valves. Table 6.2.6-2a identifies those valves that are tested during a Type C test. A list of valves that are to be Type C tested is included in the Technical Specifications. Isolation valves that are part of closed systems that are in use after a design basis event and valves that are water sealed for at least 30 days after a design basis event are not tested in the Type C test program. Table 6.2.6-2b lists the valves exempted from type C leak testing. Bases for exemptions and exceptions from type C leakage rate testing on a penetration by penetration basis are as follows:

CONTAINMENT INTEGRATED LEAK RATE TEST
TEST SUMMARY

OBJECTIVE

To verify the primary reactor containment overall integrated leakage rate is within acceptable limits.

PREREQUISITES

1. Fluid system conditions are established as applicable to simulate post accident conditions which extend the boundary of the Containment Building.

2. Containment pressure retaining boundary, leakage limiting boundary, and isolation valve leak tests have been satisfactorily performed.

3. All containment isolation valves have been closed by normal actuation methods.

4. The vendor containment over pressurization (structural integrity) test has been successfully completed.

TEST METHOD

1. Perform the containment integrated leak rate test per Appendix J to 10 CFR Part 50.

2. Perform the leakage rate calculation by using the mass-point methodology as described by ANSI/ANS 56.8-~~1981, ANSI N45.4-1972, and BN-TOP-1.~~ 1987.

3. If during the performance of a Type A test, excessive leakage occurs through locally testable penetrations or isolation valves, these leakage paths may be isolated and the Type A test continued until completion. The sum of the post repaired minimum pathway local leakage rate values will be added to the UCL. ~~per ANSI/ANS 56.8-1981. 1987.~~

4. Verify proper operation of containment narrow and wide range pressure instrumentation and alarms used in conjunction with post accident monitoring.

ACCEPTANCE CRITERIA

1. The Containment Integrated Leak Rate Test meets the requirements of Appendix J to 10 CFR Part 50.

Note: The containment structural integrity test described in FSAR Section 3.8 may be performed concurrently with the Integrated Leak Rate Test.

2. The containment pressure instrumentation operates in accordance with design, as described in FSAR Section 7.5.

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TABLE 6.2.6-3

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CONTAINMENT VESSEL PRESSURE AND LEAK TEST
REACTOR BUILDING CONTAINMENT PENETRATION STATUS

Penetration	Description	Status
	A. PENETRATION STATUS DURING TEST PERFORMANCE	
X-1	Equipment Hatch	Closed
X-2A	Elevation 719'- 4" Air Lock	Closed
X-2B	Elevation 760'- 4" Air Lock	Closed
X-3	Fuel Transfer Tube	Closed
X-4	Heating and Ventilating Air Flow	Vented
X-5	Heating and Ventilating Air Flow	Vented
X-6	Heating and Ventilating Air Flow	Vented
X-7	Heating and Ventilating Air Flow	Vented
X-8	Seal Welded Spare	Vented ~~Closed~~ (See Note 1)
X-8A	Feedwater Bypass	Normal Lineup
X-8B	Feedwater Bypass	Normal Lineup
X-8C	Feedwater Bypass	Normal Lineup
X-8D	Feedwater Bypass	Normal Lineup
X-9A	Heating and Ventilating Air Flow	Vented
X-9B	Heating and Ventilating Air Flow	Vented
X-10A	Heating and Ventilating Air Flow	Vented
~~X-10B~~	~~Heating and Ventilating Air Flow~~	~~Vented~~
X-11	Heating and Ventilating Air Flow	Vented
X-12A	Feedwater System	Normal Lineup
X-12B	Feedwater System	Normal Lineup
X-12C	Feedwater System	Normal Lineup
X-12D	Feedwater System	Normal Lineup
X-13A	Main and Reheat Steam System	Normal Lineup
X-13B	Main and Reheat Steam System	Normal Lineup
X-13C	Main and Reheat Steam System	Normal Lineup
X-13D	Main and Reheat Steam System	Normal Lineup
X-14A	Steam Generator Blowdown System	Normal Lineup
X-14B	Steam Generator Blowdown System	Normal Lineup
X-14C	Steam Generator Blowdown System	Normal Lineup
X-14D	Steam Generator Blowdown System	Normal Lineup
X-15	Chemical and Volume Control System	Drained & Vented
X-16	Chemical and Volume Control System	Normal Lineup
X-17	Residual Heat Removal System	Normal Lineup
X-18	Seal Welded Spare	Vented (See Note 1)
X-19A	Safety Injection System	Normal Lineup
X-19B	Safety Injection System	Normal Lineup
X-20A	Safety Injection System	Normal Lineup

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CONTAINMENT VESSEL PRESSURE AND LEAK TEST
REACTOR BUILDING CONTAINMENT PENETRATION STATUS

Penetration	Description	Status
X-20B	Safety Injection System	Normal Lineup
X-21	Safety Injection System	Normal Lineup
X-22	Safety Injection System	Normal Lineup
X-23	PAS Cont. Air Intk LC Tr-B	Vented
X-24	Reactor Coolant System	Normal Lineup
X-25A	Radiation System	Drained & Vented
X-25B	Containment ΔP Sensor (PdT30-42)	Vented
X-25C	Containment ΔP Sensor (PdT30-30c)	Vented
X-25D	Radiation Sampling System	Drained & Vented
X-26A	ILRT Sensor Line	In Use (See Note 2)
X-26B	ILRT Sensor Line	In Use (See Note 2)
X-26C	Containment ΔP Sensor (PdT30-43)	Vented
X-27A	Radiation Sampling System	~~Drained & Vented~~ Normal Lineup
X-27B	Radiation Sampling System	~~Drained & Vented~~ Normal Lineup
X-27C	Radiation Sampling System	~~Drained & Vented~~ Normal Lineup
X-27D	Radiation Sampling System	~~Drained & Vented~~ Normal Lineup
X-28	PAS Cont. Sump Return Tr-B	Drained & Vented
X-29	Component Cooling System	Drained & Vented
X-30	Safety Injection System	Drained & Vented
X-31	Fire Protection	Drained & Vented
X-32	Safety Injection System	Normal Lineup
X-33	Safety Injection System	Normal Lineup
X-34	Control Air System	Vented ~~(See Note~~ JST 6-20-94
X-35	Component Cooling Water System	Drained & Vented
X-36	SG Chem. Cleaning	Vented (See Note 1)
X-37	Maintenance Port	Vented (See Note 1)
X-38	Seal Welded Spare	Vented (See Note 1)
X-39A	Waste Disposal System	Vented
X-39B	Waste Disposal System	Vented
X-39C	Seal Welded Spare	Vented (See Note 1)
X-39D	Seal Welded Spare	Vented (See Note 1)
X-40A	Auxiliary Feedwater System	Normal Lineup
X-40B	Auxiliary Feedwater System	Normal Lineup
X-40C	Seal Welded Spare	Vented (See Note 1)
X-40D	Hydrogen Purge	Vented
X-41	Waste Disposal System	~~Normal Lineup~~ Drained & Vented
X-42	Primary Water System	Drained & Vented
X-43A	Chemical and Volume Control System	Normal Lineup
X-43B	Chemical and Volume Control System	Normal Lineup
X-43C	Chemical and Volume Control System	Normal Lineup
X-43D	Chemical and Volume Control System	Normal Lineup
X-44	Chemical and Volume Control System	Drained & Vented
X-45	Waste Disposal System	Vented
X-46	Waste Disposal System	Drained and Vented

TABLE 6.2.6-3 1097 S00 PKG

CONTAINMENT VESSEL PRESSURE AND LEAK TEST
REACTOR BUILDING CONTAINMENT PENETRATION STATUS

Penetration	Description	Status
X-47A	Ice Condenser System	~~Normal Lineup~~ In Use
X-47B	Ice Condenser System	~~Normal Lineup~~ In Use
X-48A	Containment Spray System	Normal Lineup
X-48B	Containment Spray System	Normal Lineup
X-49A	Containment Spray System	Normal Lineup
X-49B	Containment Spray System	Normal Lineup
X-50A	Component Cooling System	Drained & Vented
X-50B	Component Cooling System	Drained & Vented
X-51	Seal Welded Spare	Vented (See Note 1)
X-52	Component Cooling System	Drained & Vented
X-53	Component Cooling System	Drained & Vented
X-54	Thimble Renewal	In Use
X-55	Seal Welded Spare	Vented (See Note 1)
X-56A	Essential Raw Cooling Water	Drained & Vented
X-56B	Seal Welded Spare	Vented (See Note 1)
X-57A	Essential Raw Cooling Water	Drained & Vented
X-57B	Seal Welded Spare	Vented (See Note 1)
X-58A	Essential Raw Cooling Water	Drained & Vented
X-58B	~~Seal Welded Spare~~ RCS Pressure Sensor	~~Vented~~ Normal Lineup
X-59A	Essential Raw Cooling Water	Drained & Vented
X-59B	Seal Welded Spare	Vented (See Note 1)
X-60A	Essential Raw Cooling Water	Drained & Vented
X-60B	Seal Welded Spare	Vented (See Note 1)
X-61A	Essential Raw Cooling Water	Drained & Vented
X-61B	Seal Welded Spare	Vented (See Note 1)
X-62A	Essential Raw Cooling Water	Drained & Vented
X-62B	Seal Welded Spare	Vented (See Note 1)
X-63A	Essential Raw Cooling Water	Drained & Vented
X-63B	Seal Welded Spare	Vented (See Note 1)
X-64	Air Conditioning System	Drained & Vented
X-65	Air Conditioning System	Drained & Vented
X-66	Air Conditioning System	Drained & Vented
X-67	Air Conditioning System	Drained & Vented
X-68	Essential Raw Cooling Water	Drained & Vented
X-69	Essential Raw Cooling Water	Drained & Vented
X-70	Essential Raw Cooling Water	Drained & Vented
X-71	Essential Raw Cooling Water	Drained & Vented
X-72	Essential Raw Cooling Water	Drained & Vented
X-73	Essential Raw Cooling Water	Drained & Vented
X-74	Essential Raw Cooling Water	Drained & Vented
X-75	Essential Raw Cooling Water	Drained & Vented
X-76	Control and Service Air System	Vented
X-77	Demineralized Water and Cask Decon	Drained & Vented
X-78	Fire Protection	Drained & Vented
X-79A	Ice Blowing	Vented
X-79B	Negative Return	Vented

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TABLE 6.2.6-3

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CONTAINMENT VESSEL PRESSURE AND LEAK TEST
REACTOR BUILDING CONTAINMENT PENETRATION STATUS

Penetration	Description	Status
X-80	Heating and Ventilating Air Flow	Vented
X-81	Waste Disposal System	Drained & Vented
X-82	Fuel Pool Cooling and Cleaning System	Drained & Vented
X-83	Fuel Pool Cooling and Cleaning System	Drained & Vented
X-84A	Radiation Sampling System	Drained & Vented
X-84B	RVLIS	~~Vented~~ Normal Lineup
X-84C	RVLIS	~~Vented~~ Normal Lineup
X-84D	RVLIS	~~Vented~~ Normal Lineup
X-85A	Radiation Sampling System	Drained & Vented
X-85B	Radiation Sampling System	Drained & Vented
X-85C	Containment ΔP Sensor (PdT30-45)	Vented
X-85D	Seal Welded Spare	Vented (See Note 1)
X-86A	PAS Cont. Air Intk UC Tr-A	Vented
X-86B	PAS Cont. Air Rtrn Tr-A	Vented
X-86C	PAS Cont. Sump Rtrn Tr-A	Drained & Vented
X-86D	Seal Welded Spare	Vented (See Note 1)
X-87A	Seal Welded Spare	~~Drained &~~ Vented (See Note 1)
X-87B	RVLIS	~~Vented~~ Normal Lineup
X-87C	RVLIS	~~Vented~~ Normal Lineup
X-87D	RVLIS	~~Vented~~ Normal Lineup
X-88	Seal Welded Spare	Vented (See Note 1)
X-89	Seal Welded Spare	Vented (See Note 1)
X-90	Control Air System Tr-B	Vented
X-91	Control Air System Tr-A	Vented
X-92A	Hydrogen Analyzer Tr-B	Vented
X-92B	Hydrogen Analyzer Tr-B	Vented
X-92C	PAS Hot Leg 1 Tr-A	Drained & Vented
X-92D	Seal Welded Spare	Vented (See Note 1)
X-93	Radiation Sampling System	Drained & Vented
X-94A	Seal Welded Spare	Vented (See Note 1)
X-94B	Radiation Monitoring System	Vented
X-94C	Radiation Monitoring System	Vented
X-95A	Seal Welded Spare	Vented (See Note 1)
X-95B	Radiation Monitoring System	Vented
X-95C	Radiation Monitoring System	Vented
X-96A	ILRT Sensor Line	In Use (See Note 2)
X-96B	ILRT Sensor Line	In Use (See Note 2)
X-96C	Containment ΔP Sensor (Pdt 30-44)	Vented
X-97	Containment ~~Pressure~~ (PdT30-133)	Vented
X-98	Seal Welded Spare	Vented (See Note 1)

ΔP Sensor

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TABLE 6.2.6-3

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CONTAINMENT VESSEL PRESSURE AND LEAK TEST
REACTOR BUILDING CONTAINMENT PENETRATION STATUS

Penetration	Description	Status
X-99	Hydrogen Analyzer Tr-A	Vented
X-100	Hydrogen Analyzer Tr-A	Vented
X-101	Seal Welded Spare	Vented (See Note 1)
X-102	Seal Welded Spare	Vented (See Note 1)
X-103	Seal Welded Spare	Vented (See Note 1)
X-104	Seal Welded Spare	Vented (See Note 1)
X-105	PAS Cont. Air Rtrn Tr-B	Vented
X-106	PAS Hot Leg 3 Tr-B	Drained & Vented
X-107	Residual Heat Removal System	Normal Lineup
X-108	Maintenance Port	Vented (See Note 1)
X-109	Maintenance Port	Vented (See Note 1)
X-110	Seal Welded Spare	Vented (See Note 1)
X-111	Seal Welded Spare	Vented (See Note 1)
X-112	Seal Welded Spare	Vented (See Note 1)
X-113	Seal Welded Spare	Vented (See Note 1)
X-114	Ice Condenser System	~~Normal Lineup~~ In Use
X-115	Ice Condenser System	~~Normal Lineup~~ In Use
X-116	Seal Welded Spare	Vented (See Note 1)
X-117	Maintenance Port	Vented (See Note 1)
X-118	Layup Water Treatment	Stet In Use SST 6-22-94
X-119	Seal Welded Spare	Vented ~~Vented~~ (See Note 1)
X-120	Seal Welded Spare	Vented ~~Vented~~ (See Note 1)
X-121E	Electrical Penetration	Vented (See Note ~~1~~) SST 6-22-94
X-122E	Electrical Penetration	Vented (See Note ~~1~~) SST 6-22-94
X-123E	Electrical Penetration	Vented (See Note ~~1~~)
X-124E	Electrical Penetration	Vented "
X-125E	Electrical Penetration	Vented "
X-126E	Electrical Penetration	Vented "
X-127E	Electrical Penetration	Vented "
X-128E	Electrical Penetration	Vented "
X-129E	Electrical Penetration	Vented "
X-130E	Electrical Penetration	Vented "
X-131E	Electrical Penetration	Vented "
X-132E	Electrical Penetration	Vented "
X-133E	Electrical Penetration	Vented "
X-134E	Electrical Penetration	Vented "
X-135E	Electrical Penetration	Vented "
X-136E	Electrical Penetration	Vented "
X-137E	Electrical Penetration	Vented "
X-138E	Electrical Penetration	Vented "
X-139E	Electrical Penetration	Vented "
X-140E	Electrical Penetration	Vented "
X-141E	Electrical Penetration	Vented " SST 6-22-94
X-142E	Electrical Penetration	Vented (See Note ~~1~~)

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TABLE 6.2.6-3

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CONTAINMENT VESSEL PRESSURE AND LEAK TEST
REACTOR BUILDING CONTAINMENT PENETRATION STATUS

Penetration	Description	Status
X-143E	Electrical Penetration	Vented (See Note ~~3~~ 1) JBT 6-22-94
X-144E	Electrical Penetration	Vented "
X-145E	Electrical Penetration	Vented "
X-146E	Electrical Penetration	Vented "
X-147E	Electrical Penetration	Vented "
X-148E	Electrical Penetration	Vented "
X-149E	Electrical Penetration	Vented "
X-150E	Electrical Penetration	Vented "
X-151E	Electrical Penetration	Vented "
X-152E	Electrical Penetration	Vented "
X-153E	Electrical Penetration	Vented "
X-154E	Electrical Penetration	Vented "
X-155E	Electrical Penetration	Vented "
X-156E	Electrical Penetration	Vented "
X-157E	Electrical Penetration	Vented "
X-158E	Electrical Penetration	Vented "
X-159E	Electrical Penetration	Vented "
X-160E	Electrical Penetration	Vented "
~~X-161E~~	~~Electrical Penetration~~	~~Vented "~~
X-162E	Seal Welded Spare	Vented (See Note 1)
X-163E	Electrical Penetration	Vented (See Note ~~3~~ 1) JBT 6-22-94
X-164E	Electrical Penetration	Vented "
X-165E	Electrical Penetration	Vented "
X-166E	Electrical Penetration	Vented "
X-167E	Electrical Penetration	Vented "
X-168E	Electrical Penetration	Vented "
X-169E	Electrical Penetration	Vented "
X-170E	Electrical Penetration	Vented "
X-171F	Electrical Penetration	Vented "
X-172E	Electrical Penetration	Vented "
X-173E	Electrical Penetration	Vented "
X-174E	Electrical Penetration	Vented "

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TABLE 6.2.6-3 (Con't) 1097 S00 PKG

CONTAINMENT VESSEL PRESSURE AND LEAK TEST
REACTOR BUILDING CONTAINMENT PENETRATION STATUS

Penetration	Description	Status
	B. TESTABLE PENETRATIONS REQUIRED TO BE INSERVICE DURING TEST PERFORMANCE	
X-26A X-26B	Integrated Leak Rate Test	Isolation valves required to be open to monitor containment pressure (See Note 2)
~~X-41~~	~~Waste Disposal System~~	~~Floor and equipment sump discharge required to handle any unanticipated liquid leakage during testing~~
X-47A	Ice Condenser System	Glycol cooling supply to air handling units in ice condenser required to ensure ice condition is maintained
X-47B	Ice Condenser System	Same as X-47A
X-54	Thimble Renewal	Used as pressurization point for air compressors
X-96A X-96B	Integrated Leak Rate Test	Isolation valves required to be open to monitor containment pressure (See Note 2)

 Added by Amendment 52

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TABLE 6.2.6-3

CONTAINMENT VESSEL PRESSURE AND LEAK TEST
REACTOR BUILDING CONTAINMENT PENETRATION STATUS (cont'd)

Penetration	Description	Status
X-107	Residual Heat Removal System	Residual heat removal system required inservice to remove decay heat from fuel
X-114	Ice Condenser System	Glycol return from air handling units required to ensure ice condition is maintained
~~X-115~~	~~Ice Condenser~~ System	Same as X-114
X-118	Hatch	Used as source for verification flow and post-test depressurization; opened during DBE event to drain water from annulus to Reactor Building floor and equipment drain sump

Add

Notes:

1. These penetrations are closed. Venting is provided by the design of the penetration such that any leakage is detectable by the integrated leak rate test.

2. These penetrations are designed to facilitate ILRT performance. It may not be necessary to utilize all of the penetrations. If not in use, the penetration is vented.

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT UNITS 1 AND 2

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
FSAR CHAPTER 3, AMENDMENT 79