SIMULATOR CERTIFICATION

FORM 474

QUADRENNIAL REPORT

Table of Contents

Plant E. I. Hatch

Docket # 50-321 & 50-366

Georgia Power Company

The analysis of documents referenced in this submittal was completed prior to the date of submittal as listed on NRC Form 474 for each unit. The revision number listed for these documents in the attachments to the submittal and in the documentation retained by GPC in support of this submittal are the revision levels used in the Certifications process.

As procedure revisions Technical Specification ammendments, and revisions to other documents used in the certification process occur, they will be reviewed in accordance with the Plant E. I. Hatch training department procedures (including the Simulator Certification Program) to determine any required action.

NRC FORM 474
(10 89)
10 CFR § 55.45(b)
55.4 and 55.5

U.S. NUCLEAR REGULATORY COMMISSION
Approved by OMB
3150-0138
Expires 10-31-89

SIMULATION FACILITY CERTIFICATION

INSTRUCTIONS. This form is to be filed for initial certification, recertification (if required), and for any change to a simulation facility performance testing plan made after initial submittal of such a plan. Provide the following information, and check the appropriate box to indicate reason for submittal.

FACILITY
E. I. HATCH, Unit 2, Nuclear Power Station

DOCKET NUMBER
50- 366

LICENSEE
Georgia Power Company

DATE
4-19-94

This is to certify that: 1. the above named facility licensee is using a simulation facility consisting solely of a plant-referenced simulator that meets the requirements of 10 CFR § 55.45; 2. the simulation facility meets the guidance contained in ANSI/ANS 3.5, 1985, as endorsed by NRC Regulatory Guide 1.149; and 3. documentation is available for NRC review in accordance with 10 CFR § 55.45(b). If there are any exceptions to the certification of item 2 above, check here ☐ and describe fully on additional pages as necessary.

NAME *(or other identification)* AND LOCATION OF SIMULATION FACILITY
E. I. Hatch Simulator, US 1 North, Approx. 12 miles north of Baxley, Ga. Zip 31513

XX SIMULATION FACILITY PERFORMANCE TEST ABSTRACTS ATTACHED. *(For performance tests conducted in the period ending with the date of this certification)*

DESCRIPTION OF PERFORMANCE TESTING COMPLETED *(Attach additional page(s) as necessary, and identify the item description being continued)*

See Attachment 2: Performance Tests Completed

XX SIMULATION FACILITY PERFORMANCE TESTING SCHEDULE ATTACHED. *(For the conduct of approximately 25% of performance tests per year for the four year period commencing with the date of this certification.)*

DESCRIPTION OF PERFORMANCE TESTING TO BE CONDUCTED *(Attach additional page(s) as necessary, and identify the item description being continued)*

See Attachment 3: Performance Test Schedule

XX PERFORMANCE TESTING PLAN CHANGE *(For any modification to a performance testing plan submitted on a previous certification)*

DESCRIPTION OF PERFORMANCE TESTING PLAN CHANGE *(Attach additional page(s) as necessary, and identify the item description being continued)*

See Attachment 3: Changes Since Last Report

☐ RECERTIFICATION *(Describe corrective actions taken, attach results of completed performance testing in accordance with 10 CFR § 55.45(b)(5)(v). Attach additional page(s) as necessary, and identify the item description being continued.)*

Any false statement or omission in this document, including attachments, may be subject to civil and criminal sanctions. I certify under penalty of perjury that the information in this document and attachments is true and correct.

SIGNATURE – AUTHORIZED REPRESENTATIVE	TITLE	DATE
Lewis Sumner	Nuclear Plant General Manager	4-19-94

In accordance with 10 CFR § 55.5, Communications, this form shall be submitted to the NRC as follows:

BY MAIL ADDRESSED TO: Director, Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

BY DELIVERY IN PERSON TO THE NRC OFFICE AT: 7920 Norfolk Avenue
Bethesda, MD

Attachment 1

EXCEPTIONS

04/14/94

Plant E. I. Hatch, Unit 2

Docket # 50-366

Georgia Power Company

LISTING OF CERTIFICATION EXCEPTIONS

NO.	TESTS	ANSI SPEC	TITLE OF EXCEPTION
001	05-01	3.1.2(1)(a)	Loss of Coolant; Significant PWR Steam Generator Leaks
002	05-28	3.1.2(18)	Failure of Reactor Coolant Pressure and Volume Control Systems (PWR)
003	06-07	3.1.1 (7)	Startup, Shutdown and Power Operations with Less Than Full Reactor Coolant Flow
004	06-12	3.1.1 (9c)	Core Performance: Reactivity Coefficient Measurements
005	06-13	3.1.1 (9d)	Core Performance; Control Rod Worth Using Permanently Installed Instrumentation
006	06-11	3.1.1 (9b)	Core Performance: Determination of Shutdown Margin
007	07-XX	B.1.2.1,2,3	Appendix B: Transient Performance
008	ALL	3.3.1	Degree of Panel Simulation
009	05-15	3.1.2 (5)	Loss of Condenser Level Control
01[illegible]	06-15,16, and 17	App.B B1.1	Steady State Performance

PERFORMED BY: Rutan DATE: 01/31/89 REVIEWED BY: J. Richter DATE: 04/14/94

APPROVED BY: R. S. Grantham OPERATIONS TRAINING SUPERVISOR DATE: 04/18/94

EXCEPTION NUMBER: 001

ANSI SPECIFICATION: 3.1.2(1)(a) TITLE: Loss of Coolant; Significant PWR Steam Generator Leaks (05-01) :

EXCEPTION: Unit One and Unit Two

GPC takes exception to this requirement. There is no Performance Test which will be performed as part of the Plant Hatch simulator certifi- process.

JUSTIFICATION:

Plant Hatch is a General Electric Boiling Water Reactor (BWR). This requirement is not applicable to the Hatch simulator.

TITLE:	PLANT E. I. HATCH SIMULATOR CERTIFICATION PLAN CERTIFICATION EXCEPTIONS REV: 1

PERFORMED BY: Rutan DATE: 01/31/89 REVIEWED BY: J. Richter DATE: 04/14/94

APPROVED BY: R. S. Grantham OPERATIONS TRAINING SUPERVISOR DATE: 04/18/94

EXCEPTION NUMBER: 002

ANSI SPECIFICATION: 3.1.2(18) TITLE: Failure of Reactor Coolant Pressure and Volume Control Systems (PWR) (05-28)

EXCEPTION: Unit One and Unit Two

GPC takes exception to this requirement. There is no Performance Test which will be performed as part of the Plant Hatch simulator certification process.

JUSTIFICATION:

Plant Hatch is a General Electric Boiling Water Reactor (BWR). This requirement is not applicable to the ' tch simulator.

PERFORMED BY: Rutan DATE: 02/06/89 REVIEWED BY: J. Richter DATE: 04/14/94

APPROVED BY: R. S. Grantham OPERATIONS TRAINING SUPERVISOR DATE: 04/18/94

EXCEPTION NUMBER: 003

ANSI SPECIFICATION: 3.1.1(7) TITLE: Startup, Shutdown and Power Operations with Less than Reactor Coolant Flow (06-07)

EXCEPTION: Unit One and Unit Two

GPC takes exception to this requirement. This requirement is satisfied by performance of ANSI specifications 3.1.1 (1), (2), (8a) and (8b); therefore, this test will not be performed as a separate test.

JUSTIFICATION:

Plant Hatch is a General Electric Boiling Water Reactor (BWR-4) and can only startup and shutdown with less than full reactor coolant flow. Performance of tests 06-01, 06-02, 06-08 and 06-09 satisfy the above mentioned ANSI 3.1.1 specifications and will be performed in lieu of this requirement.

PERFORMED BY: Rutan DATE: 02/06/89 REVIEWED BY: J. Richter DATE: 04/14/94

APPROVED BY: R. S. Grantham OPERATIONS TRAINING SUPERVISOR DATE: 04/18/94

EXCEPTION NUMBER: 004

ANSI SPECIFICATION: 3.1.1(9c) TITLE: Core Performance; Reactivity Coefficient Measurements (06-12)

EXCEPTION: Unit One and Unit Two

GPC takes exception to this requirement. There is no Performance Test which will be performed as part of the Plant Hatch simulator certification process.

JUSTIFICATION:

Plant Hatch personnel do not perform this type of testing on the plant nor in the simulator; therefore, this test is considered to be not applicable to Plant Hatch.

PERFORMED BY: Rutan DATE: 02/06/89 REVIEWED BY: J. Richter DATE: 04/14/94

APPROVED BY: R. S. Grantham OPERATIONS TRAINING SUPERVISOR DATE: 04/18/94

EXCEPTION NUMBER: 005

ANSI SPECIFICATION: 3.1.1(9d) TITLE: Core Performance; Control Rod Worth Using Permanently Installed Istrumentation (06-13)

EXCEPTION: Unit One and Unit Two

GPC takes exception to this requirement. There is no Performance Test which will be performed as part of the Plant Hatch simulator certification process.

JUSTIFICATION:

Plant Hatch personnel do not perform this type of testing on the plant nor in the simulator; therefore, this test is considered to be not applicable to Plant Hatch.

TITLE:	PLANT E. I. HATCH SIMULATOR CERTIFICATION PLAN CERTIFICATION EXCEPTIONS REV: 1

PERFORMED BY: Rutan DATE: 03/29/89 REVIEWED BY: J. Richter DATE: 04/14/94

APPROVED BY: R. S. Grantham OPERATIONS TRAINING SUPERVISOR DATE: 04/18/94

EXCEPTION NUMBER: 006

ANSI SPECIFICATION: 3.1.1(9b) TITLE: Core Performance; Determination of Shutdown Margin (06-11)

EXCEPTION: Unit One and Unit Two

GPC takes exception to this requirement. There is no Performance Test which will be performed as part of the Plant Hatch simulator certification process.

JUSTIFICATION:

Reactor Engineering performs the Shutdown Margin Determination at the time of initial core criticality by mean of a "In-Sequence Criticality Verification". Reactor Engineering personnel do not utilize the simulator to train on Shutdown Margin Determination; therefore, this test is considered to be not applicable to Plant Hatch.

TITLE: PLANT E. I. HATCH SIMULATOR CERTIFICATION PLAN
CERTIFICATION EXCEPTIONS REV: 1

PERFORMED BY: Rutan DATE: 03/29/89 REVIEWED BY: J. Richter DATE: 04/14/94

APPROVED BY: R. S. Grantham OPERATIONS TRAINING SUPERVISOR DATE: 04/18/94

EXCEPTION NUMBER: 007

ANSI SPECIFICATION: B.1.2.1,2,3 TITLE: Appendix B: Transient Performance (07-XX)

XCEPTION: Unit One and Unit Two

This section of Appendix B requires that all Appendix B listed transient tests be recorded with a resolution of 0.5 second or less. GPC takes exception to this requirement of recording at 0.5 second criteria. All transient tests were recorded at 1.0 second intervals.

JUSTIFICATION:

Current data recording capabilities meet the requirement of Section 4.4, Monitoring Capability, of the standard. GPC's recording program has the ability to record at 0.5 second intervals but the plant reference data recording program (Safety Parameter Display System) utilizes a 1.0 second recording interval. GPC has determined that to better compare plant reference data to simulator response the 1.0 second recording interval was warranted for the following reasons:

1. System dynamic response time for most parameters is approximately equal to one second. Plant recorders have a typical response time interval of one to five seconds.
2. System response items that require less than one second intervals are logical items (i.e., annunciators, setpoint actuations, etc.) that were observed during performance testing. These items are not a part of the plotted information.

PERFORMED BY: Rutan DATE: 04/10/89 REVIEWED BY: J. Richter DATE: 04/14/94

APPROVED BY: R. S. Grantham OPERATIONS TRAINING SUPERVISOR DATE: 04/18/94

EXCEPTION NUMBER: 008

ANSI SPECIFICATION: 3.2.1 TITLE: Degree of Panel Simulation :

EXCEPTION: Unit One and Unit Two

This section requires that the simulator shall contain sufficient operational panels to provide the control, instrumentation, alarms, and other man-machine interfaces to conduct the normal plant evolutions of 3.1.1 and respond to the malfunctions of 3.1.2. GPC takes exception to this requirement for control room panel features:

PANEL(S)	DESCRIPTION OF CONTROL ROOM PANEL FEATURE (REMOTE FUNC. NO.)
P609 P611	MSIV Low Vacuum Trip Bypass switches 2A71-S34A,B,C,D (#134)
P662	Condensate Booster Pump Minimum flow Control Valve Controller (#151)
P662	Reactor Feedwater Pump "A" Minimum Flow Control Valve Controller (#137)
P662	Reactor Feedwater Pump "B" Minimum Flow Control Valve Controller (#200)
P663	EHC Cabinet 400 Hz and Electrical Malfunction reset switch (#144)
P621	RCIC valve F022 (Test line to CST) automatic closure bypass link AA-14 (# 235)
P609 P611	Reactor Recirc. System RPT Breaker Trip Bypass switches (#171,172)
P609 P611	Main Turbine Control Valve Testing jumpers (#173 thru 176)
P609 P611	Main Turbine Stop Valve Testing jumpers (#258 thru 269)

JUSTIFICATION:

GPC considers the above listed hardware exceptions to be warranted due to the following reasons:

- o These panels are not located in the simulator since they are not used frequently enough by the control room operator in the course of plant operation to justify the additional cost to the simulator. On-the-Job training is considered to be adequate for these items.
- o Each of these panel features are software simulated and controlled by the simulator instructor via remote function operation.

PERFORMED BY: Shaffer DATE: 08/15/89 REVIEWED BY: J. Richter DATE: 04/14/94

APPROVED BY: R. S. Grantham OPERATIONS TRAINING SUPERVISOR DATE: 04/18/94

EXCEPTION NUMBER: 009

ANSI SPECIFICATION: 3.1.2(5) TITLE: Loss of Condenser Vacuum Including Loss of Condenser Level Control

EXCEPTION: Unit One and Unit Two

GPC takes exception to this requirement only with regard to the Loss of Condenser Level Control portion. A Loss of Condenser Level Control test will not be performed as part of the Plant Hatch simulator certification process.

JUSTIFICATION:

A failure of the condenser hotwell level control system transient would be slow in developing and would cause no major changes in associated plant critical parameters without ample warning and in sufficient time to allow for malfunction correction prior to any major plant transient occurring.

There are no JTA tasks selected for training in the simulator associated with condenser hotwell level control system malfunctions; therefore, the Loss of Condenser Level Control portion of this specification is considered not applicable to Plant Hatch.

PERFORMED BY: Rutan & Shaffer DATE: 12/05/89 REVIEWED BY: J. Richter DATE: 04/14/94

APPROVED BY: R. S. Grantham OPERATIONS TRAINING SUPERVISOR DATE: 04/18/94

EXCEPTION NUMBER: 012

ANSI SPECIFICATION: App.B_B1.1_ TITLE: Steady State Performance

EXCEPTION: Unit One and Unit Two

Appendix B of the Standard requires Reactor Water Cleanup return flow to be a recorded parameter for steady state performance of a minimum of three points over the power range. GPC takes exception to this requirement to the extent there is no Performance Test with a recorded Reactor Water Cleanup return flow as part of the Plant Hatch simulator certification process.

JUSTIFICATION:

Reactor Water Cleanup return flow is not a parameter that is monitored in the Reference Plant Control Room.

Plant E. I. Hatch, Unit 2 Docket # 50-366 Georgia Power Company

Attachment 1 04/14/94

Software and hardware deficiencies noted during the Control Room Environment review and during performance of ANSI 3.1.1 and 3.1.2 testing are listed below. Corrective action for each deficiency is to modify the simulator as necessary to meet the ANSI 3.5 acceptance criteria. Projected dates of completion for each of the deficiencies is also listed below by its Simulator Change Request (SCR) number.

SCR Number	Anticipated Completion	Software Discrepancies
8812025	05/95	Torus temperature response on SRV lift
8910040	05/95	Drywell pressure 100 seconds after LOCA
9103002	05/95	Drywell temperature appears to be a function of saturation pressure
9103003	05/95	Drywell temperature spread constants cause incorrect temp response.
9208014	08/94	Rx pressure goes very low on a loss of feedwater test 0702
9307007	05/95	Drywell pressure doesn't increase enough initially upon malf 123.
9404004	06/94	Simulator not keeping up with real time in test 08-01 section B.

Attachment 2

PERFORMANCE TESTS COMPLETED

04/14/94

Plant E. I. Hatch, Unit 2

Docket # 50-366

Georgia Power Company

ABSTRACTS

The Performance Test Lists attached with this submittal provide a cross reference between the title of the performance test, the test number assigned by Georgia Power, and the specific ANSI/ANS 3.5 requirement the test addresses. The Performance Test Abstracts included in this attachment provide a description of the specific tests performed to show compliance with ANSI/ANS 3.5 - 1985 and Regulatory Guide 1.149.

The simulator malfunctions tested for certification are listed on the Performance Test Abstracts. Malfunctions listed as tested at a particular severity are variable severity malfunctions and specify the severity used for the test. The range of severity available for all variable malfunctions is 1 to 100 percent. The exact parametric value (e.g., 100 gpm) of the tested severity is also listed in the abstract.

The variable severity malfunctions will also have a specified ramp rate used in the test. The ramp rate is the insertion rate (rate of increase in severity) associated with a variable severity malfunction. The maximum ramp rate available is 1,000 percent per minute.

The baseline data used in the evaluation of simulator response to the test is specified on the abstract. Items typically used for comparison include the Final Safety Analysis Report (FSAR), Reference Plant strip charts, and Safety Parameter Display System (SPDS) tapes. Additionally, an Abnormal Event Analysis was utilized as baseline data for the Unit One evaluation, as applicable. The Abnormal Event Analysis (NEDE - 30156) is a training document developed by the General Electric Company for use in Transient and Accident Analysis training. This document uses analyses specific to a BWR-4 with steam turbine driven feedwater pumps. These analyses are collected from various sources, including the FSAR, the reload analyses for Plant Hatch, the Emergency Procedure Guidelines and other related studies. The FSAR data was only used to verify the proper direction of the parameters evaluated.

The data collected for each performance test was analyzed by a group of individuals knowledgeable of Plant Hatch systems and operations. These individuals are listed on the test abstract as Panel Members and are listed below specifying the qualifications of each individual. Differing opinions (if any) expressed by the panel members related to the performance of the simulator are also listed on the Performance Test Abstract.

PANEL MEMBERS LISTED ON TEST ABSTRACTS

NAME	POSITION	QUALIFICATIONS
B. Smith	Plant Instructor - Operations Nuclear	NRC SRO Instructor Certification (BWR) Bachelor of Science Nuclear Engineering Technology
S. Loesch	Senior Engineer I	Masters Degree Mechanical Engineering
S. Stone	Shift Supervisor	SRO License at Plant E. I. Hatch
A. Wolfe	Shift Supervisor Training	SRO License at Plant E. I. Hatch
J. Kelly	Plant Instructor - Operations Nuclear	SRO License at Plant E. I. Hatch

Performance Tests

The following operability tests will be conducted annually (Computer Real Time and Simulator Operating Limits Tests and Appendix B - Steady State and Transient Performance Tests):

615 Steady State Performance 25% power

616 Steady State Performance at 75% power

617 Steady State Performance at 100% rated thermal power and verify stability for 60 mi nutes

701 Transient Performance; Manual Scram

702 Transient Performance; Simultaneous trip of all feedwater pumps

703 Transient Performance; Simultaneous closure of all Main Steam Isolation Valves

704 Transient Performance; Simultaneous trip of all recirculation pumps

705 Transient Performance; Single recirculation pump trip

706 Main Turbine Trip (From the max power level which will not result in an immediate R x scram)

707 Transient Performance; Maximum power ramp (master recirc controller in manual) down to approximately 75% and back up to 100%

708 Transient Performance; Maximum size reactor coolant system rupture combined with a loss of offsite power

709 Transient Performance; Maximum size unisolable main steam line rupture

710 Transient Performance; Simultaneous closure of all Main Steam Isolation Valves comb ined with single stuck open safety/relief valves (inhibit activation of high pressu e Emergency Core Cooling Systems)

801 Computer Real Time Test

802 Simulator Operating Limits

PERFORMANCE TEST NUMBER: 615-1 **Date Tested:** 03/18/93

TITLE: Steady State Performance 25% power

Initial Conditions: 25% rated power, 37% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Steady State Conditions

Termination Criteria: After 5 consecutive OD-3 options 2 & 3 printouts 1 minute apa rt have been collected.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions): 10 12

Unit 1 Baseline Data: SCR 1-88-5 Dated 5/20/88 - A615A

34GO-OPS-005-1S - Attachments 1 & 2

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 615-2 **Date Tested:** 03/18/93

TITLE: Steady State Performance at 25%

Initial Conditions: 25% rated power, 37% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Steady State Conditions

Termination Criteria: After 5 consecutive OD-3 options 2 & 3 printouts 1 minute apa
rt have been collected.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions): 12

Unit 1 Baseline Data: Best Estimate Judgment

Unit 2 Baseline Data: Tape Dated 3/23/88 - F615A

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 616-1 **Date Tested:** 03/18/93

TITLE: Steady State Performance at 75% power

Initial Conditions: 75% rated power, 64% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Steady State Conditions

Termination Criteria: After 5 Consecutive OD-3 options 2 & 3 printouts 1 minute apa rt have been collected.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions): 10 12

Unit 1 Baseline Data: SPDS Tape 500000024, SCR 1-86-9 Dated 11/22/86 - A616A

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 616-2 **Date Tested:** 03/18/93

TITLE: Steady State Performance at 75%

Initial Conditions: 75% rated power, 64% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Steady State Conditions

Termination Criteria: After 5 Conecutive OD-3 options 2 & 3 printouts 1 minute apar t have been collected.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions): 12

Unit 1 Baseline Data: Best Estimate Judgment

Unit 2 Baseline Data: SCR 2-88-5 Dated 4/17/88 - B616A

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 617-1 **Date Tested:** 03/18/93

TITLE: Steady State Performance at 100% rated thermal power and verify stability fo r 60 minutes

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Steady State Conditions

Termination Criteria: 60 Minutes

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions): 10 12

Unit 1 Baseline Data: SPDS Tape 500000023, SCR 1-87-1 Dated 1/1/87 - C617A

SPDS Tape 500000031, SCR 1-88-6 Dated 9/4/88 - E617A

SCR 1-87-5 Dated 7/23/87

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 617-2 **Date Tested:** 03/18/93

TITLE: Steady State Performance at 100% rated thermal power and verify stability fo r 60 minutes

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Steady State Conditions

Termination Criteria: 60 minutes

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions): 12

Unit 1 Baseline Data: Best Estimate Judgment

Unit 2 Baseline Data: SPDS Tape 500000027, SCR 2-87-1 Dated 1/26/87
SCR 2-88-10 Dated 8/5/88

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 701-0 **Date Tested:** 10/05/93

TITLE: Transient Performance; Manual Scram

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Manual Reactor Scram by placing the Reactor Mode Switch to Shu tdown

Termination Criteria: Reactor Water Level has been restored to above +32 inches and the EHC system has stabilized Reactor pressure.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: SPDS Tape 500000024, SCR 1-86-9 Dated 11/22/86 - C701C

Scram Report 1-87-4

Strip Charts - 1B21-R613, 1B21-623A, 1B31-R614, 1C51-R603A, 1C32-R607
Strip Charts - 1C32-R608, 1C32-R609, 1N40-R903

Unit 2 Baseline Data: SCR 2-88-5 Dated 4/17/88

Panel Members: B. Smith, J. Kelly, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 702-0 **Date Tested:** 10/05/93

TITLE. Transient Performance; Simultaneous trip of all feedwater pumps

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #87A & 87B (Feedwater pump Trip)

Termination Criteria: Reactor water level restored to above +32 inches by HPCI and RCIC

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Any SPDS Tape that show a loss of feedwater transient

AEA (NEDE - 30156)

FSAR 14.3.4.3

Unit 2 Baseline Data: SCR 2-88-10 Dated 8/5/88

SCR 2-87-2 Dated 4/22/87; SCR 208806 Dated on 5/27/88

Strip Charts - 2C32-R607, 2C32-R609

FSAR 15.1.18

Panel Members: B. Smith, J. Kelly, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 703-0 **Date Tested:** 10/05/93

TITLE: Transient Performance; Simultaneous closure of all Main Steam Isolation Valv
es

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #127A, 127B, 127C and 127D (MSIVs Fail Shut)

Termination Criteria: Reactor water level has been restored to above +32 inches.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: AEA (NEDE - 30156); FSAR 14.3.1.3

SCR 1-86-2

Strip Charts - 1B31-R614, 1N40-R903, 1B21-R613

Strip Charts - 1C51-R603A, 1C32-R608, 1B21-R623A, 1C32-R609, 1C32-R607.

Unit 2 Baseline Data: SPDS Tape - 500000006; SCR 2-86-6 Dated 8/29/86 - B703B

SPDS Tape - 500000027; SCR 2-87-1 Dated 1/26/87 - D703B

FSAR 15.1.4

Strip Chart 232-R609

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 704-0 **Date Tested:** 10/06/93

TITLE: Transient Performance; Simultaneous trip of all recirculation pumps

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Malfunctions 37A and 37B (Recirculation Pump Drive Motor Break er Trip)

Termination Criteria: Reactor feedwater temperature stabilizes

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: AEA (NEDE - 30156)

FSAR 14.3.5.3

Unit 2 Baseline Data: FSAR 15.1.22

SCR 2-88-12

Strip Charts - 2C51-R603A, 2C32-R608, 2B21-R623A, 2C32-R609, 2C32-R607
Strip Charts - 2B31-R614, 2B31-R613

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 705-0 **Date Tested:** 10/06/93

TITLE: Transient Performance; Single recirculation pump trip

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #38(B) (Recirculation Pump Field Breaker Trip)

Termination Criteria: Reactor feedwater temperature stabilizes

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Strip Charts - 1C51-R603A, 1C32-R608, 1B21-R623A, 1C32-R609, 1C32-R607
Strip Charts - 1B31-R614, 1B21-R613

AEA (NEDE - 30156)

Unit 2 Baseline Data: Strip Charts - 2C51-R603A, 2C32-R608, 2B21-R623A, 2C32-R609, 2B31-R614
Strip Charts - 2B21-R613, 2C32-R607

SCR 2-87-7

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 706-0 **Date Tested:** 10/06/93

TITLE: Main Turbine Trip (From the max power level which will not result in an imme diate Rx scram)

Initial Conditions: 30% rated power, 40% rated core flow, MOL, Equilibrium Xe.

Transient Initiator: Malfunction #122 (Main Turbine Trip)

Termination Criteria: Reactor water level restored to above +32 inches by Reactor F eedwater Pumps following a high reactor pressure scram

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: AEA (NEDE - 30156 4.2)

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B. Smith, J. Kelly, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 707-0 **Date Tested:** 10/06/93

TITLE: Transient Performance; Maximum power ramp (master recirc controller in manual) down to approximately 75% and back up to 100%

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Reduce the output of the Recirc Master Controller

Termination Criteria: Reactor water level has stabilized following the power increase.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgment

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 708-1 **Date Tested:** 10/06/93

TITLE: Transient Performance; Maximum size reactor coolant system rupture combined with a loss of offsite power

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malf #210 (Unit 1 DBA LOCA-Recirc Suction Double Ended Break) at max. ramp rate with Malf #161 (Loss of Offsite Powe

Termination Criteria: Approximately 5 minutes

Deficiencies Unit 1: 8910040 Drywell pressure after LOCA @ 100 seconds???
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: AEA (NEDE - 30156 11.2, 13.4.1)

FSAR 14.4.3

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 708-2 **Date Tested:** 10/06/93

TITLE: Transient Performance; Maximum size reactor coolant system rupture combined with a loss of offsite power

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malf #35 (Unit 2 DBA LOCA-Recirc Discharge Double Ended Break) at max. ramp rate with Malf #161 (Loss of offsite Pow

Termination Criteria: Approximately 5 minutes

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgment

Unit 2 Baseline Data: FSAR 15.1.39

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 709-0 **Date Tested:** 10/06/93

TITLE: Transient Performance; Maximum size unisolable main steam line rupture

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #123 (Steam Line Break - Complete Severance of the "B" Main Steam Line in the Drywell)

Termination Criteria: RWL has been raised to above +32" and Low Pressure Emergency Core Cooling Systems are injecting into the Reactor Ve

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: FSAR 14.4.5

AEA (NEDE 30156 11.2.2.2)

Unit 2 Baseline Data: FSAR 15.1.40

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 710-0 **Date Tested:** 10/06/93

TITLE: Transient Performance; Simultaneous closure of all Main Steam Isolation Valves combined with single stuck open safety/relief valves (inhibit activation of high pressure Emergency Core Cooling Systems)

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malf #127A, 127B, 127C & 127D (Outboard MSIVs Fail Shut), Malf #130A (SRV Fails Open), and Malf #104 (HPCI Turbine Trip)

Termination Criteria: RWL has been restored to above +32" and the condensate & CBPs trip on low condenser hotwell level

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: AEA (NEDE 30156, 11.3.6.7, 11.3.6.8)

FSAR 14.3.4.2

Unit 2 Baseline Data: FSAR 15.1.17

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PREPARED BY: S. Loesch DATE:04/13/94 REVIEWED BY: J. Richter DATE:04/14/94

SIMULATOR TEST NUMBER: 08-01 TITLE: Simulator Real Time Test ANSI App. A3.1

SECTION A: STEADY STATE REAL TIME TEST

Date Tested: 04/13/94

Initial Conditions:
Part One - 100% rated power
Part Two - 50% rated power
Part Three - 25% rated power

Evolution Initiator: At any terminal connected to CCC85A computer start the program "realtime". This program compares model scheduled time to actual model run time. A message is sent to terminal after five minutes. Message "test is done" is received when results are acceptable. A message "not keeping up with real time" is received when results are not acceptable.

Termination Criteria: Five minutes.

Deficiencies: None

Exceptions: None

SECTION B: TRANSIENT REAL TIME TEST

Date Tested: 04/13/94

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator:
Part One - Malfunction # 87A (Feedwater Pump Trip).
Part Two - Malfunction # 178 (4KV bus 2A fault) and # 179 (4KV bus 2B fault)
Part Three - Malfunction # 161 (Loss of Offsite Power) and # 210 Recirc loop suction break) at 100% severity (DBA LOCA)

Evolution Initiator: At any terminal connected to CCC85A computer start the program "realtime". This program compares model scheduled time to actual model run time. A message is sent to terminal after five minutes. Message "test is done" is received when results are acceptable. A message "not keeping up with real time" is received when results are not acceptable.

Termination Criteria: Five minutes.

Deficiencies: SCR # 9404004

Exceptions: None

PREPARED BY: S. Loesch DATE: 04/13/94 REVIEWED BY: J. Richter DATE: 04/14/94

SIMULATOR TEST NUMBER: 08-02 TITLE: Simulator Operating Limits Testing ANSI Section 4.3

PART ONE

Date Tested: 04/13/94	Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.
Transient Initiator:	Bring Reactor pressure to it's Engineering Limit by any method desired. When Reactor pressure increases to the Engineering Limit (1325 psig), verify the simulator automatically goes into freeze and a message appears on the computer console. Place the simulator back into the RUN mode and bring reactor pressure above the Model Limit (1410 psig).
Termination Criteria:	When Reactor pressure is brought above the Model Limit (1410 psig), the simulator automatically goes into freeze and can not be placed into the RUN mode.
Deficiencies:	None
Exceptions:	None

PART TWO

Date Tested: 04/13/94	Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.
Transient Initiator:	Bring Containment Pressure to it's Engineering Limit by any method desired. When Containment Pressure increases to the Engineering Limit (62.5 psig), verify the simulator automatically goes into freeze and a message appears on the computer console. Place the simulator back into the RUN mode and bring Containment Pressure above the Model Limit (100 psig).
Termination Criteria:	When Containment Pressure is brought above the Model Limit (100 psig), the simulator automatically goes into freeze and can not be placed into the RUN mode.
Deficiencies:	None
Exceptions:	None

PREPARED BY: S. Loesch DATE: 04/13/94 REVIEWED BY: J. Richter DATE: 04/13/94

SIMULATOR TEST NUMBER: 08-02 TITLE: Simulator Operating Limits Testing ANSI Section 4.3

PART THREE

Date Tested: 04/13/94	Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.
Transient Initiator:	Bring Fuel Cladding Temperatures indicating gross fuel failures to it's Engineering Limit by any method desired. When Fuel Cladding Temperatures indicating gross fuel failures increases to the Engineering Limit (2200 ^{O}F), verify the simulator automatically goes into freeze and a message appears on the computer console. Place the simulator back into the RUN mode and bring Fuel Cladding Temperatures above the Model Limit (2800 ^{O}F).
Termination Criteria:	When Fuel Cladding Temperatures are brought above the Model Limit (2800 ^{O}F), the simulator automatically goes into freeze and can not be placed into the RUN mode.
Deficiencies:	None
Exceptions:	None

PART FOUR

Date Tested: 04/13/94	Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.
Transient Initiator:	Bring Gross Voiding in the Core to it's Engineering Limit by any method desired. When Gross Voiding in the Core increases to the Engineering and Model Limit (-308 inches of Reactor Water Level), verify the simulator automatically goes into freeze and a message appears on the computer console. Attempt to place the simulator back into the RUN mode.
Termination Criteria:	After Gross Voiding in the Core has been brought to it's Engineering and Model Limit (-308 inches of Reactor Water Level), the simulator can not be placed into the RUN mode.
Deficiencies:	None
Exceptions:	None

PREPARED BY: S. Loesch DATE: 04/13/94 REVIEWED BY: J. Richter DATE: 04/14/94

SIMULATOR TEST NUMBER: 08-02 TITLE: Simulator Operating Limits Testing ANSI Section 4.3

PART FIVE

Date Tested: 04/13/94	Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.
Transient Initiator:	Bring Suppression Pool High Temperature to it's Engineering Limit by any method desired. When Suppression Pool High Temperature increases to the Engineering Limit (Temperature > Tsat -20 ^{O}F), verify the simulator automatically goes into freeze and a message appears on the computer console. Place the simulator back into the RUN mode and bring Suppression Pool High Temperature to it's Model Limit (Temperature > Tsat - 10 ^{O} F).
Termination Criteria:	When Suppression Pool High Temperature is brought to it's Model Limit (Temperature > Tsat -10 ^{O} F), the simulator automatically goes into freeze and can not be place into the RUN mode.
Deficiencies:	None
Exceptions:	None

Performance Tests

The following performance tests will be conducted (in addition to the annual operability tests) in the first year following certification (Normal Operation Tests, Malfunction Tests, and Transient Tests not tested as Annual Operability Tests):

502 Loss of coolant; Inside primary containment

505 Loss of Coolant; Small reactor coolant breaks including demonstration of saturation condition

507 Loss of instrument air to the extent that the whole system or individual headers ca n lose pressure and affect the plant's static or dynamic performance

521 Loss of protective system channel

525 Turbine Trip

526 Generator Trip

532 Main feed line break inside containment

534 Nuclear instrumentation failure (s)

538 Reactor pressure control system failure including turbine bypass failure (BWR)

604 Reactor trip followed by recovery to rated power

PERFORMANCE TEST NUMBER: 502-0 **Date Tested:** 10/11/93

TITLE: Loss of coolant; Inside primary containment

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #48 (Intermediate break in Drywell) at 50% severit y (2125 gpm) at a maximum ramp rate.

Termination Criteria: Low pressure ECCS systems have raised reactor water level abo ve +60 inches.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: AEA (NEDE - 30156)

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B. Smith, J. Kelly, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 505-1 **Date Tested:** 10/11/93

TITLE: Loss of Coolant; Small reactor coolant breaks including demonstration of satur ation condition

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Malfunction #54 (Coolant Leakage in Drywell - 2G31-F001 weld l eakage) at 100% severity (30 gpm) at a maximum ramp rate

Termination Criteria: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Deficiencies Unit 1: 9310004
0
0
0

Deficiencies Unit 2: 9310004
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: AEA (NEDE - 30156)

Scram Report 1-85-5

Strip Chart data for Drywell, Torus, and SBGT parameters

Best Estimate Judgment for other parameters

Unit 2 Baseline Data: All parameters from 0000 12/13/87 to 2400 on 12/14/87

FSAR 15.1.32

Strip Chart data for Drywell Pressure and SBGT Flow parameter s

Best Estimate Judgment for other parameters

Panel Members: S.Stone, B. Smith

Differing Opinion: None

PERFORMANCE TEST NUMBER: 505-2 **Date Tested:** 10/11/93

TITLE: Loss of coolant; Small reactor coolant breaks including demonstratio of sat uration condition

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Malfunction #215 (Reactor Level Instrument Line "B" Leak - ref erence leg serverance)

Termination Criteria: Reactor water level has been restored to 32" by HPCI and RCIC

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: SCR 1-87-1

Unit 2 Baseline Data: Plant Strip Chart data on initial portion of transient for AP RM
Best Estimate Judgement

2-88-1

Panel Members: B. Smith, J. KELLY, S. LOESCH

Differing Opinion: None

PERFORMANCE TEST NUMBER: 507-0 **Date Tested:** 10/11/93

TITLE: Loss of instrument air to the extent that the whole system or individual hea ders can lose pressure and affect the plant's static or dynamic performance

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilbrium Xe.

Transient Initiator: Malfunction #191 (Instrument Air Leakage). This malf. will cau se a total loss of service air & nonessential instr.

Termination Criteria: Reactor water level has been restored to above +32 inches

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

FASR 15.1.28

Panel Members: B. Smith, J. Kelly, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 521-0 **Date Tested:** 10/11/93

TITLE: Loss of protective system channel

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #57A (Loss of Normal Power to Reactor Protective C
hannel)

Termination Criteria: Malfunction insertion time of 5 minutes

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Strip Charts 1C51-R603A, R603C; 1C32-R607, R608, R609; 1B2
1-R615
Strip Chart 1B31-R614

Unit 2 Baseline Data: Strip Charts 2C51-R603A; 2C32-R607, R608, R609; 2B31-R614

Strip Chart 2B21-R615

Panel Members: B. Smith, J. Kelly, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 525-0 **Date Tested:** 10/11/93

TITLE: Turbine Trip

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #122 (Turbine Trip).

Termination Criteria: Reactor water level restored to above +32 inches by the React or Feedwater Pumps.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: SPDS Tape 500000023 (A525A), SCR 1-87-1

Strip Charts 1B31-R614, 1C32-R607, 1C32-R608, 1C32-R609

Strip Charts 1B21-R615, 1B21-R623A, 1C51-R603A

Scram Report 1-88-4

Unit 2 Baseline Data: Strip Charts 2C51-R603A, 2C32-R608, 2B21-R623A, 2B21-R615, 2C32-R609
Strip Charts 2C32-R607, 2B31-R614

Scram Report 2-85-6

Panel Members: B. Smith, J. Kelly, S. Losech

Differing Opinion: None

PERFORMANCE TEST NUMBER: 526-0 **Date Tested:** 10/11/93

TITLE: Generator Trip

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #209 (Generator Load Rejection)

Termination Criteria: Reactor water level restored to above +32 inches by Reactor F eedwater Pumps.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: SPDA Tape 500000026 (A526A)

AEA (NEDE 30156) REF DATA: S
cram Report 1-88-1
FSAR 14.3.1.1

Strip Charts 1C51-R603A; 1C32-R607, R608, R609; 1B21-R615, R6
23A; 1B31-R614

Unit 2 Baseline Data: Scram Report 2-86-4

FSAR 15.1.1

Strip Charts 2C51-R603A; 2C32-R607, R608, R609; 2B31-R614;
2B21-R623A, R615

Panel Members: B. Smith, J. Kelly, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 532-0 **Date Tested:** 10/11/93

TITLE: Main feed line break inside containment

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium xe.

Transient Initiator: Malfunction #229 (Feedwater Line "A" Break Inside Containment) at 100% severity (Double Ended Break) at a max. ramp rate

Termination Criteria: HPCI has restored reactor water level to above +32 inches

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: AEA (NEDE - 30156)

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B. Smith, S. Loesch, J. Kelly

Differing Opinion: none

PERFORMANCE TEST NUMBER: 534-1 **Date Tested:** 10/12/93

TITLE: Nuclear instrumentation failure (s)

Initial Conditions: Reactor Startup in progress, BOL, Reactor Mode Switch in Startu p, 172°F Reactor coolant temperature

Transient Initiator: Malfunction #3A (SRM Failure - Upscale) and then Malfunction # 4C, (SRM Failure - Downscale)

Termination Criteria: Malfunction #4C has been inserted for 1 minute

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Plant Procedures, Drawings and Surveillances

Best Estimate Judgement

Unit 2 Baseline Data: Plant Procedures, Drawings and Surveillances

Best Estimate Judgement

Panel Members: B. Smith, S. Loesch, J. Kelly

Differing Opinion: None

PERFORMANCE TEST NUMBER: 534-2 **Date Tested:** 10/12/93

TITLE: Nuclear instrumentation failures (s)

Initial Conditions: Rx Power about 10%, Reactor Startup in progress, BOL, Rx Mode S witch in Startup, 925 psig Rx pressure

Transient Initiator: Malfunction #7A, (IRM Failure - Inoperative) and then Malfunct ion #9C (IRM Failure - Upscale)

Termination Criteria: Malfunction #9C has been inserted for 1 minute.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Plant Procedures, Drawings and Surveillances

Unit 2 Baseline Data: Plant Procedures, Drawings and Surveillances

Panel Members: B. Smith, S. Loesch, J. Kelly

Differing Opinion: None

PERFORMANCE TEST NUMBER: 534-3 **Date Tested:** 10/11/93

TITLE: Nuclear instrumentation failure(s)

Initial Conditions: 100% rated power, 100% rated core flow, EOL Equilibrium Xe

Transient Initiator: Malfunction #11A (APRM Failure - Upscale) and then Malfunction #14C, (APRM Failure - Inoperative)

Termination Criteria: After Malfunction #14C response has been verified.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Plant Procedures, Drawings and Surveillances

Unit 2 Baseline Data: Plant Procedures, Drawings and Surveillances

Panel Members: B. Smith, J. Kelly, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 534-4 **Date Tested:** 10/11/93

TITLE: Nuclear Instrumentation Failure(s)

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Malfunction #15B, (RBM Failure - Upscale) and then Malfunction #17A, (RBM Failure - Inoperative)

Termination Criteria: After Malfunction #17A resonse has been verified.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Plant Procedures, Drawings and Surveillances

Best Estimate Judgement

Unit 2 Baseline Data: Plant Procedures, Drawings and Surveillances

Best Estimate Judgement

Panel Members: B. Smith, S. Loesch, J. Kelly

Differing Opinion: None

PERFORMANCE TEST NUMBER: 534-5 **Date Tested:** 10/11/93

TITLE: Nuclear Instrumentation failures(s)

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Malfunction #200D (Recirculation Loop Flow Instrument Failure) & then Malfunction #20J, Rev. 0 (LPRM Failure-Downscale)

Termination Criteria: After Malfunction #20J response has been verified.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Plant Procedures, Drawings and Surveillances

Best Estimate Judgement

Unit 2 Baseline Data: plant Procedures, Drawings and Surveillances

Best Estimate Judgement

Panel Members: B. Smith, S. Loesch, J. Kelly

Differing Opinion: None

PERFORMANCE TEST NUMBER: 538-1 **Date Tested:** 10/12/93

TITLE: Reactor pressure control system failure including turbine bypass failure (BW R)

Initial Conditions: 75% rated power, MOL

Transient Initiator: Malfunction #137A (EHC Pressure Regulator Fails High)

Termination Criteria: Reactor Water Level has been restored to above +32 inches by RCIC and HPCI

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: FSAR 14.3.4.1

AEA (NEDE - 30156)

Best Estimate Judgement

Unit 2 Baseline Data: FSAR 15.1.5

Best Estimate Judgement

Panel Members: B. Smith, S. Loesch, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 538-2 **Date Tested:** 10/11/93

TITLE: Reactor pressure control system failure including turbine bypass failure (BW R)

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #138A (EHC pressure Regulator Fails Low)

Termination Criteria: Main Generator MWe Stabilized

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: AEA (NEDE - 30156)

Best Estimate Judgement

Unit 2 Baseline Data: FSAR 15.1.6

Best Estimate Judgement

Panel Members: B. Smith, S. Loesch, J. Kelly

Differing Opinion: None

PERFORMANCE TEST NUMBER: 538-3 **Date Tested:** 10/11/93

TITLE: Reactor pressure control system failure including turbine bypass failure (BW R)

Initial Conditions: 100% rated power, 100% rated core flow, EOL

Transient Initiator: Manual Scram with Malfunction #134 (All Bypass Valves Fail Clo sed)

Termination Criteria: Low Low Set has actuated and suppression pool temperature beg ins to increase

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, S. Loesch, J. Kelly

Differing Opinion: None

PERFORMANCE TEST NUMBER: 604-0 **Date Tested:** 12/06/90

TITLE: Reactor trip followed by recovery to rated power

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Following a Reactor Scram from at power, perform a reactor sta rtup per 34GO-OPS-001, "Normal Reactor Startup".

Termination Criteria: The plant has been returned to rated Reactor Power.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions): 00 0 8

Unit 1 Baseline Data: SPDS Tape 500000024

Scram Report 1-86-9 on 11/22/86 (A604A1, 2, 4, 5)

Unit 2 Baseline Data: 34GO-OPS-001-2S

Data package Sheets 7-42, Rev. 8

Panel Members: B. Smith, S. Loesch, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 604-0 **Date Tested:** 12/06/90

TITLE: Reactor Trip followed by recovery to rated power

Initial Conditions:

Transient Initiator: See page 1, this page is only for 1 interval (mode switch to r un to rated power)

Termination Criteria:

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgment

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members:

Differing Opinion:

Performance Tests

The following performance tests will be conducted (in addition to the annual operability tests) in the second year following certification (Normal Operation Tests, Malfunction Tests, and Transient Tests not tested as Annual Operability Tests):

503 Loss of coolant; Outside Primary Containment

506 Failure of safety and relief valves

508 Loss or degraded electrical power to the station, including loss of offsite power

515 Loss of condenser vacuum including loss of condenser level control

516 Loss of service water or cooling to individual components

522 Control Rod failure including stuck rods, uncoupled rods, drifting rods, rod drops, and misaligned rods

523 Inability to drive control rods

524 Fuel cladding failure resulting in high activity in reactor coolant or off gas and the associated high radiation alarms

527 Failure in automatic control system(s) that affect reactivity and core heat removal

608 Plant shutdown from rated power to hot standby

609 Cooldown to cold shutdown conditions

PERFORMANCE TEST NUMBER: 503-1 **Date Tested:** 10/12/93

TITLE: Loss of coolant; Outside Primary Containment

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Malfunction #52 (Reactor water clean-up system leak on check v alve F122) at 5% severity (12.5 gpm) at a max. ramp rate

Termination Criteria: Five minutes after malfunction insertion

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgment

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 503-2 **Date Tested:** 10/12/93

TITLE: Loss of coolant; Outside primary containment

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Malf. #52 (Rx water clean-up system leak on check valve F122) severity increased to 50% (125 gpm) at a max. ramp rate

Termination Criteria: Upon RWCU system isolation

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgment

Unit 2 Bas‹line Data: Best Estimate Judgment

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 503-3 **Date Tested:** 10/12/93

TITLE: Loss of coolant; Outside primary containment

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Malfunction #126 (Steam leak in the turbine building) 12.6%

Termination Criteria: Upon PCIS group I isolation (MSIV) closure)

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgment

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 506-0 **Date Tested:** 10/12/93

TITLE: Failure of safety and relief valves

Initial Conditi . 100% rated power, 100% rated core flow, EOL, Equilibirium Xe

Transient Initiator: Malfunction #130-A (Main Steam Relief Valve Fails Full Open)

Termination Criteria: Suppression Pool water level is increasing with Suppression P ool temperature >100 degrees

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Unit 1 Baseline Data on Strip Charts from 10/23/87 (Event Fil e 87-067)

Unit 2 Baseline Data: Unit 2 Baseline Data in test file(s) B506A (Event File 88-038).

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 508-0 **Date Tested:** 10/31/91

TITLE: Loss or degraded electrical power to the station, including loss of offsite power

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Malfunction #161 (Loss of Off-Site Power)

Termination Criteria: Malfunction insertion for 3 minutes

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 515-0 **Date Tested:** 10/12/93

TITLE: Loss of condenser vacuum including loss of condenser level control

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #70 (Circulating Waater System Leak) at 100% severity (100,000 gpm) at a maximum ramp rate.

Termination Criteria: Reactor water level has been restore to above +32 inches.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, S. Loesch, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 516-1 **Date Tested:** 10/13/93

TITLE: Loss of service water or cooling to individual components

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium, Xe.

Transient Initiator: Malfunction #67B (Service Water Pump Trip)

Termination Criteria: Plant Service Water pressure and Flow have returned to normal

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 516-2 **Date Tested:** 10/13/93

TITLE: Loss of service water or cooling to individual components

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #67A, 67C, 67D (Service Water Pump Trip)

Termination Criteria: Reactor water level has been restored to above +32 inches.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 522-1 **Date Tested:** 10/13/93

TITLE: Control Rod failure including stuck rods, uncoupled rods, drifting rods, rod drops, and misaligned rods

Initial Conditions: 52% rated power, 55% rated core flow, EOL.

Transient Initiator: Malfunction #22 (Control Rod XX-YY Failure - Stuck).

Termination Criteria: Stall Flow has been verified in both the insert and withdraw directions

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: none

PERFORMANCE TEST NUMBER: 522-2 **Date Tested:** 10/13/93

TITLE: Control rod failure including stuck rods, uncoupled rods, drifting rods, rod drops and misaligned rods

Initial Conditions: 52% rated power, 55% rated core flow, EOL.

Transient Initiator: Malf#23 inserted on the rod selected for Part 1 & removal of Malf#22 after the selected CRD Mech. has been withdrawn

Termination Criteria: Reactor power has stabilized after the control rod drop

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement on Abnormal Event Analysis (NEDE 30156)

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 522-3 **Date Tested:** 10/13/93

TITLE: Control rod failure including stuck rods, uncoupled rods, drifting rods, rod d rops and misaligned rods

Initial Conditions: 52% rated power, 55% rated core flow, EOL.

Transient Initiator: Malfunction #25 (Control Rod XX-YY Failure - Drift In)

Termination Criteria: The selected control rod has gone to the full in overtravel p osition after opening the cooling water supply valve.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgment

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 522-4 **Date Tested:** 10/13/93

TITLE: Control rod failure including stuck rods, uncoupled rods, drifting rods, rod drops, and misaligned rods

Initial Conditions: 52% rated power, 55% rated core flow, EOL.

Transient Initiator: Malfunction #26 (Control Rod XX-YY Failure - Scram).

Termination Criteria: Reactor pressure and power have stabilized following the individual rod scram.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 523-0 **Date Tested:** 10/13/93

TITLE: Inability to drive control rods

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: With Malfunction #30A (Control Rod Drive System Pump Trip) inserted, attempt to move a control rod.

Termination Criteria: Verification that the control rod will not move.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 524-1 **Date Tested:** 10/13/93

TITLE: Fuel cladding failure resulting in high activity in reactor coolant or off gas and the associated high radiation alarms

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Malfunction #189 (Fuel Cladding Failure/Gross Fuel Failure) at 15% severity (9 Million R/hr) at a maximum ramp rate.

Termination Criteria: Pretreatment Radiation level increases

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgment

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B. Smith, S. Loesch, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 524-2 **Date Tested:** 10/13/93

TITLE: Fuel Cladding failure resulting high activity in reactor coolant or off gas and the associated high radiation alarms

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibirum Xe.

Transient Initiator: Malfunction #189 (Fuel Cladding Failure/Gross Fuel Failure) at 15% severity (9 Million R/hr) at a maximum ramp rate.

Termination Criteria: Primary Containment radiation levels stabilize following the scram and PCIS Group I isolation

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, S. Loesch, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 527-0 **Date Tested:** 10/13/93

TITLE: Failure in automatic control system(s) that affect reactivity and core heat removal

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #103 (HPCI Inadvertent Start-up)

Termination Criteria: HPCI trip on high reactor water level.

Deficiencies Unit 1: 8910007
0
0
0

Deficiencies Unit 2: 8910007
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement utilizing Abnormal Event Analysis (NE DE 30156)

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, S. Loesch, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 608-0 **Date Tested:** 10/31/91

TITLE: Plant shutdown from rated power to hot standby

Initial Conditions: 99% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Shutdown the Reactor Plant to Hot Standby per 34GO-OPS-013-2, Normal Plant Shutdown.

Termination Criteria: The reactor mode switch has been placed to Startup and HOT St andby position

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, A. Wolfe

Differing Opinion: None

PERFORMANCE TEST NUMBER: 609-0 **Date Tested:** 10/31/91

TITLE: Cooldown to cold shutdown conditions

Initial Conditions: 2.4% rated power, 18% rated core flow, 920 psig reactor pressur e, MOL.

Transient Initiator: Perform a Plant Cooldown per 34GO-OPS-013, Plant Cooldown

Termination Criteria: Cold shutdown conditions have been established

Deficiencies Unit 1: 8910048
0
0
0

Deficiencies Unit 2: 8910048
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: SCR 1-87-3

Unit 2 Baseline Data: SCR 2-87-5

Panel Members: B. Smith, A. Wolfe

Differing Opinion: None

Performance Tests

The following performance tests will be conducted (in addition to the annual operability tests) in the third year following certification (Normal Operation Tests, Malfunction Tests, and Transient Tests not tested as Annual Operability Tests):

509 Loss of Degraded Electrical Power to Station, Loss of Emergency Power

510 Loss or Degraded Electrical Power to the Station, Loss of Emergency Generators.

511 Loss or degraded electrical power to the station, loss of power to the plant's elec trical distribution buses.

531 Main steam line break outside containment

533 Main feed line break outside containment

601 Plant startup - Cold to hot standby. The starting conditions shall be cold shutdow n conditions of temperature and Pressure. Removal of the reactor vessel head is not a required condition for simulation.

602 Nuclear startup from hot standby to rated power

603 Turbine startup and generator synchronization

605 Operations at hot standby

606 Load Changes

610 Core performance testing; Heat Balance

PERFORMANCE TEST NUMBER: 509-0 **Date Tested:** 03/05/93

TITLE: Loss of Degraded Electrical Power to Station, Loss of Emergency Power

Initial Conditions: 100% rated power, 100% rated core flow, Equilibrium Xe.

Transient Initiator: Malfunction # 182 (4KV Emergency Bus 2E Fault)

Termination Criteria: Reactor water level restored to above +32 inches

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Drawings and Procedures and Best estimate Judgement

Unit 2 Baseline Data: Drawings and Procedures and Best estimate judgement

Panel Members: B. Smith

Differing Opinion: None

PERFORMANCE TEST NUMBER: 510-0 **Date Tested:** 03/15/93

TITLE: Loss or Degraded Electrical Power to the Station, Loss of Emergency Generators.

Initial Conditions: 100% rated power, 100% rated core flow, Equilibrium Xe.

Transient Initiator: Malfunction #62C (Diesel Generator Failure to Auto Start), Malfunction #184 (4KV Bus 2G Fault).

Termination Criteria: Malfunction has been inserted for 5 minutes.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Drawings and Procedures and Best Estimate Judgement

Unit 2 Baseline Data: Drawings and Procedures and Best Estimate Judgement

Panel Members: B. Smith, J. Kelley

Differing Opinion: None

TITLE:	PLANT E. I. HATCH SIMULATOR CERTIFICATION PLAN MALFUNCTION PERFORMANCE TEST ABSTRACT	REV: 2

PERFORMANCE TEST NUMBER: 511-0 **Date Tested:** 03/15/93

TITLE: Loss or degraded electrical power to the station, loss of pc wer to the plant 's electrical distribution buses.

Initial Conditions: 100% rated power, 100% rated core flow, EOL Equilibrium Xe.

Transient Initiator: Malfunction #201 (Loss of Vital AC).

Termination Criteria: Reactor water level restored to above +32 inches

Deficiencies Unit 1: 8910043 APRM FLUCTUATIONS MISSING ON RECIRC RUNBACK.
0
0
0

Deficiencies Unit 2: 8910043 APRM FLUCTUATIONS MISSING ON RECIRC RUNBACK.
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Scram Report #1-87-5 Dated 7/23/87

Unit 2 Baseline Data: Scram Report #2-87-3 Dated 7/26/87

Scram Report #2-87-4 Dated 8/3/87

SPDS Tapes from Loss of Vital AC Dated 7/26/87 or 8/3/87

Panel Members: B. Smith, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 531-0 **Date Tested:** 03/15/93

TITLE: Main steam line break outside containment

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #126A (Steam Leak in Turbine Building).

Termination Criteria: Reactor Water Level restored to above +32 inches by HPCI.

Deficiencies Unit 1: 8910008 RX PRESSURE DECREASED TOO SLOWLY ON TURBINE BUILDING BREAK.
0
0
0

Deficiencies Unit 2: 8910008 RX PRESSURE DECREASED TOO SLOWLY ON TURBINE BUILDING BREAK.
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: AEA (NEDE - 30156)

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B. Smith, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 533-0 **Date Tested:** 03/15/93

TITLE: Main feed line break outside containment

Initial Conditions: 100% rated power, 100% core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #64 (Feedwater Line "A" Break) at 100% severity the maximum ramp rate.

Termination Criteria: Reactor water level has been restored to above +32 inches by RCIC.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgment

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B. Smith, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 601-0 **Date Tested:** 03/11/93

TITLE: Plant startup - Cold to hot standby. The starting conditions shall be cold shutdown conditions of temperature and Pressure. Removal of the reactor vessel head is not a required condition for simulation.

Initial Conditions: 0% rated power, 10% rated core flow, BOL, 172°F reactor coolant temperature, Xenon free condition.

Transient Initiator: Perform a reactor startup per 34GO-OPS-001-2S, "Normal Reactor Startup".

Termination Criteria: After shell and chest warming are complete per 34GO-OPS-001-2S and the plant has been brought to 920 psig.

Deficiencies Unit 1: 8910048
9303010
0
0

Deficiencies Unit 2: 8910048
9303010
0
0

Exceptions Numbers (see attachment A for descriptions): 8

Unit 1 Baseline Data: 34GO-OPS-001-2 Data pkg shts 7-28, Rev. 5

34SV-B31-001-2 Data pkg shts 9-10, Rev. 1

Unit 2 Baseline Data: 34GO-OPS-001-2 Data pkg shts 7-28, Rev. 5

SPDS Tapes B601B1, B601B2, B601B3, D601B3, N601B3, N601B4

Tapes dated 3/17/88 to 3/26/88. Also tapes dated 12/9/86 to 12/12/86
34SV-B31-001-2 Data pkg shts 9-10, Rev. 1

Panel Members: B. Smith, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 602-0 **Date Tested:** 03/11/93

TITLE: Nuclear startup from hot standby to rated power

Initial Conditions: Reactor Critical, BOL, 920 psig reactor pressure, shell and che st warming complete, Mode Switch in Startup/Hot Standby.

Transient Initiator: Perform a reactor startup per 34GO-OPS-001-2S. "Normal Reacto r Startup".

Termination Criteria: The plant has been brought to rated power

Deficiencies Unit 1: 8910048
9303010
0
0

Deficiencies Unit 2: 8910048
9303010
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgment

Unit 2 Baseline Data: 34GO-OPS-001-2 Data pkg shts 21, 28-42 R5

SPDS Tape B602A1, D602A1, D602A2, D602A3.

Tapes dated 3/17/88 to 3/26/88. Also tapes dated 12/9/86 to 12/12/86

Panel Members: B. Smith, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 603-0 **Date Tested:** 03/11/93

TITLE: Turbine startup and generator synchronization

Initial Conditions: Reactor Critical, BOL, 920 psig reactor pressure, shell and che st warming complete, Mode Switch in Run

Transient Initiator: Perform a Main Turbine startup and a Main Generator synchroniz ation per 34GO-OPS-001, "Normal Reactor Startup".

Termination Criteria: The Main Generator is tied to the grid and carrying load

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions): 8

Unit 1 Baseline Data: Best Estimate Judgment

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B. Smith, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 605-0 **Date Tested:** 03/05/93

TITLE: Operations at hot standby

Initial Conditions: Reactor is Critical at .1% rated power, 11% rated core flow, 84 psig reactor pressure, BOL.

Transient Initiator: While in Hot Standby, Equalize around the MSIVs per 34GO-OPS-001-2, "Normal Reactor Startup".

Termination Criteria: One minute after the MSIVs have been opened and plant conditions have stabilized.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: 34GO-OPS-001-2S Data pkg sheets 24-26 R5

Unit 2 Baseline Data: 34GO-OPS-001-2S Data pkg sheets 24-26 R5

SPDS Tape B605A Dated 3/22/88

Panel Members: B. Smith, & J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 606-0 **Date Tested:** 03/11/93

TITLE: Load Changes

Initial Conditions: Nucl. Startup in prog., Rx Pwr at appx. 59%, BOL, Recirc flow ctrl in Indiv. Manual ctrl, 2nd Rx Feedwtr. pump/in serv.

Transient Initiator: Transfer Recirc Control to Master Manual Control and increase power to rated per 34GO-OPS-001-2, "Normal Reactor Startup

Termination Criteria: One minute after the reactor is at rated power and plant conditions have stabilized.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgment

Unit 2 Baseline Data: Best Estimate Judgment

Panel Members: B Smith, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 610-0 **Date Tested:** 03/17/93

TITLE: Core performance testing; Heat Balance

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Compute Core Thermal Power (Manually).

Termination Criteria: Manual heat balance data has been obtained and an OD-3 has be en printed.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: SPDS Tape 500000031, SCR 1-86-6 Dated 9/4/88 - A610B

SPDS Tape 500000023, SC.. 1-87-1 Dated 1/1/87 - C610B

Unit 2 Baseline Data: SPDS Tape 500000027, SCR 2-87-1 Dated 1/26/87 - D610B

SCR 2-88-10 Dated 8/5/88 - B610B

Panel Members: B. Smith, J. Kelley

Differing Opinion: None

Performance Tests

The following performance tests will be conducted (in addition to the annual operability tests) in the fourth year following certification (Normal Operation Tests, Malfunction Tests, and Transient Tests not tested as Annual Operability Tests):

512 Loss or degraded electical power to the station, loss of power to the individual instrumentation buses (AC) that provide power to control room indication or plant conrol functions affecting the plant's response

513 Loss or degraded electrical power to the station, loss of power to the individual instrumentation buses (DC) that provide power to control room indication or plant control functions affecting the plant's response

518 Loss of component cooling system or cooling to individual components

519 Loss of normal feedwater or normal feedwater system failure

520 Loss of all feedwater (normal and emergency)

535 Process instrumentation, alarms, and control system failures

536 Passive malfunctions in systems, such as engineered safety features, emergency feedwater systems

537 Failure of the automatic reactor trip system

614 Operator conducted surveillance testing on safety-related equipment or systems.

PERFORMANCE TEST NUMBER: 512-0 **Date Tested:** 09/28/93

TITLE: Loss or degraded electical power to the station, loss of power to the indivi dual instrumentation buses (AC) that provide power to control room indication or plant cont ol functions affecting the plant's response

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #185 (120/208V AC Instrument Bus 2A Fault)

Termination Criteria: Reactor water level is restored to above +32 inches

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Event Review Team report 92-009 (no SPDS tape available)

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 513-1 **Date Tested:** 09/29/93

TITLE: Loss or degraded electrical power to the station, loss of power to the individ ual instrumentation buses (DC) that provide power to control room indication or p lant control functions affecting the plant's response

Initial Conditions: 100% rated power, 100% rated core Flow, EOL Equilibrium Xe.

Transient Initiator: Malfunction #204 (loss of 125/250V DC Bus 2B)

Termination Criteria: After Reactor Recirc Pump 2A Flow is less than 45,200 gpm

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 513-2 **Date Tested:** 09/29/93

TITLE: Loss or degraded electrical power to the station, loss of power to the individ ual instrumentation buses (DC) that provide power to control room indication or p lant control functions affecting the plant's response

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #205 (24/48 DC Cabinet 2A Fault - 2R25-S015)

Termination Criteria: Malfunction inserted for 2 minutes

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 513-3 **Date Tested:** 09/29/93

TITLE: Loss or degraded electrical power to the station, loss of power to the individ ual instrumentation buses (DC) that provide power to control room indication or p lant control functions affecting the plant's response

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #203 (Loss of 125/250V DC Bus 2A)

Termination Criteria: Reactor water level restored to above +32 inches

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 518-1 **Date Tested:** 09/30/93

TITLE: Loss of component cooling system or cooling to individual components

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #72C (RBCCW Pump Auto Start Failure), and Malfunction #71A and 71B (RBCCW Pump Trip)

Termination Criteria: Malfunction #71A and 71B have been inserted for 5 minutes

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, S. Loesch, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 518-2 **Date Tested:** 10/01/93

TITLE: Loss of component cooling system or cooling to individual components

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Malfunction #193A and 193B, Rev 33 (Drywell Chiller Compressor Failure)

Termination Criteria: Drywell Pressure increases to 2.0 psig

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, S. Loesch, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 519-0 **Date Tested:** 10/01/93

TITLE: Loss of normal feedwater or normal feedwater system failure

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malfunction #91B (Loss of Extraction Steam to 4th Stage Feedwa ter Heater)

Termination Criteria: Feedwater inlet temperature decreases and Reactor Power stabi lizes

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement utilizing FSAR date

Abnormal Event Analysis data (NEDE 30156) for all other param eters

Unit 2 Baseline Data: Best Estimate Judgement utilizing FSAR data

Panel Members: B. Smith, S. Loesch, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 520-0 **Date Tested:** 10/05/93

TITLE: Loss of all feedwater (normal and emergency)

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malf #113 (RCIC Auto Isolates), #108 (HPCI Auto Isolates), 30A (CRD 2A trips), and #80A,B,& C (CBPs Trip).

Termination Criteria: Reactor water level restored to above +32 inches.

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 535-1 **Date Tested:** 10/04/93

TITLE: Process instrumentation, alarms, and control system failures

Initial Conditions: 100% rated power, 100% rated core flow EOL, Equilibrium Xe.

Transient Initiator: Malfunction #166 (Automatic Voltage Regulator Failure)

Termination Criteria: Generator voltage and MVARs have stabilized

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

PERFORMANCE TEST NUMBER: 535-2 **Date Tested:** 10/04/93

TITLE: Process instrumentation, alarms, and control system failures

Initial Conditions: 100% rated power, 100% rated core flow, EOL Equilibrium Xe

Transient Initiator: Malfunction #213 (Torus Level Sensor Fails High)

Termination Criteria: All valves for HPCI suction transfer have repositioned to eit her fully open or closed, as required

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, J. Kelley, S. Loesch

Differing Opinion: None

TITLE:	PLANT E. I. HATCH SIMULATOR CERTIFICATION PLAN MALFUNCTION PERFORMANCE TEST ABSTRACT	REV: 2

PERFORMANCE TEST NUMBER: 535-3 **Date Tested:** 09/21/89

TITLE: Process instrumentation, alarms, and control system failures

Initial Conditions: 100% rated power, 100% rated core flow, EOL Equilibrium Xe

Transient Initiator: Deleted this test because of DCR 90-163 (Recirc Controller Replacement)

Termination Criteria: Reactor power has been returned to 100% rated thermal power

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, S. Loesch, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 535-4 **Date Tested:** 10/04/93

TITLE: Process instrumentation, alarms, and control system failures

Initial Conditions: 100% rated power, 100% raced core flow, EOL, Equilibrium Xe

Transient Initiator: Malfunction #141 (Main Turbine Lube Oil System Temperature Controller Failure)

Termination Criteria: Reactor water level is stable and above "0" inches

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, S. Loesch, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 535-5 **Date Tested:** 10/04/93

TITLE: Process instrumentation, alarms, and control system failures

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Malfunction #90 (Feedwater Master Level Controller Fails minim um)

Termination Criteria: Reactor water level is restored to above +32 inches

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: SCR 1-87-7

Unit 2 Baseline Data: SCR 2-88-10

Panel Members: B. Smith, S. Loesch, J. Kelley

Differing Opinion: None

TITLE:	PLANT E. I. HATCH SIMULATOR CERTIFICATION PLAN MALFUNCTION PERFORMANCE TEST ABSTRACT	REV: 2

PERFORMANCE TEST NUMBER: 536-0 **Date Tested:** 10/05/93

TITLE: Passive malfunctions in systems, such as engineered safety features, emergen cy feedwater systems

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe

Transient Initiator: Malfunction 87A/B (trip of both Reactor Feed pumps), #106 and #109

Termination Criteria: Reactor water level is restored to above +32 by manually cont rolling HPCI

Deficiencies Unit 1: 8909007 Recirc has small flow increase after coasting down to zero flow.
0
0
0

Deficiencies Unit 2: 8909007 Recirc has small flow increase after coasting down to zero flow.
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, S. Loesch, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 537-0 **Date Tested:** 10/05/93

TITLE: Failure of the automatic reactor trip system

Initial Conditions: 100% rated power, 100% rated core flow, EOL, Equilibrium Xe.

Transient Initiator: Malf #217 (Rx Protective System Fails to Scram on Reactor Low Level) and Malf #97 (Steam Flow Totalizer Failure -minimum

Termination Criteria: Reactor water level stabilizes

Deficiencies Unit 1: 0
0
0
0

Deficiencies Unit 2: 0
0
0
0

Exceptions Numbers (see attachment A for descriptions):

Unit 1 Baseline Data: Best Estimate Judgement

Unit 2 Baseline Data: Best Estimate Judgement

Panel Members: B. Smith, S. Loesch, J. Kelley

Differing Opinion: None

PERFORMANCE TEST NUMBER: 614-0 **Date Tested:** 08/27/93

TITLE: Operator conducted surveillance testing on safety-related equipment or systems.

Initial Conditions:

Transient Initiator:

Termination Criteria: The data package steps are complete for each of the above listed surveillances.

Deficiencies Unit 1: 8812025 All torus temp ind. increases when one SRV is opened.
0
0
0

Deficiencies Unit 2: 8812025 All torus temp ind. increases when one SRV is opened.
0
0
0

Exceptions Numbers (see attachment A for descriptions): 0 0 0 8

Unit 1 Baseline Data: Associated plant procedure acceptance criteria

Unit 2 Baseline Data: Associated plant procedure acceptance criteria

Panel Members: B. Smith, S. Loesch, J. Kelley

Differing Opinion: None

Attachment 3

PERFORMANCE TEST SCHEDULE

04/14/94

Plant E. I. Hatch, Unit 2

Docket # 50-366

Georgia Power Company

PREPARED BY: C. B. Smith DATE: 04/05/94 REVIEWED BY: J. Richter DATE: 04/06/94

APPROVED BY: R.S. GRANTHAM OPERATOR TRAINING SUPERINTENDENT DATE: 04/14/94

PERFORMANCE TESTING SCHEDULE GENERAL STATEMENT

The Performance Test Schedule includes all annual operability tests and all performance tests to be performed over a four year period. Performance tests scheduled to be completed each 12 month period includes the 15 annual operability tests and approximately 25% of the performance tests, such that each annual operability test will be performed four times over the four year period and each performance test will be performed once over the same four year period.

Performance tests will be scheduled as described. However, scheduled plant modifications may impact the Performance Test Schedule requiring unplanned delays in performing certain annual operability or performance tests. Unplanned deviations in the Performance Test Schedule will not exceed three months past the original annual due date.

PREPARED BY: C. B. Smith DATE: 04/05/94 REVIEWED BY: J. Richter DATE: 04/06/94

APPROVED BY: R.S. GRANTHAM OPERATOR TRAINING SUPERINTENDENT DATE: 04/14/94

Annual Operability Test

The following operability tests will be conducted annually (Computer Real Time and Simulator Operating Limits Tests and Appendix B - Steady State and Transient Performance Tests):

08-01	Simulator Real Time Test
08-02	Simulator Operating Limits Testing
06-15	Steady State Performance at 25% power
06-16	Steady State Performance at 75% power
06-17	Steady State Performance at 100% rated thermal power and verify stability for 60 minutes
07-01	Transient Performance; Manual scram
07-02	Transient Performance; Simultaneous trip of all feedwater pumps
07-03	Transient Performance; Simultaneous closure of all Main Steam Isolation Valves
07-04	Transient Performance; Simultaneous trip of all recirculation pumps
07-05	Transient Performance; Single recirculation pump trip
07-06	Transient Performance; Main turbine trip (from the maximum power level which will not result in an immediate reactor scram)
07-07	Transient Performance; Maximum power ramp (master recirc controller in manual) down to approximately 75% and back up to 100%
07-08	Transient Performance; Maximum size reactor coolant system rupture combined with a loss of offsite power
07-09	Transient Performance; Maximum size unisolable main steam line rupture
07-10	Transient Performance; Simultaneous closure of all Main Steam Isolation Valves combined with single stuck open safety/relief valve (inhibit activation of high pressure Emergency Core Cooling Systems)

PREPARED BY: C. B. Smith DATE: 04/05/94 REVIEWED BY: J. Richter DATE: 04/06/94

APPROVED BY: R.S. GRANTHAM OPERATOR TRAINING SUPERINTENDENT DATE: 04/14/94

Performance Tests

The following performance tests will be conducted (in addition to the annual operability tests) in the first year following certification (Normal Operation Tests, Malfunction Tests, and Transient Tests not tested as Annual Operability Tests):

05-02	Loss of coolant; Inside primary containment
05-05	Loss of coolant; Small reactor coolant breaks including demonstration of saturation condition
05-07	Loss of instrument air to the extent that the whole system or individual headers can lose pressure and affect the plant's static or dynamic performance
05-21	Loss of protective system channel
05-25	Turbine trip
05-26	Generator trip
05-32	Main feed line break inside containment
05-34	Nuclear instrumentation failure(s)
05-38	Reactor pressure control system failure including turbine bypass failure (BWR)
06-04	Reactor trip followed by recovery to rated power;

PREPARED BY: C. B. Smith DATE: 04/05/94 REVIEWED BY: J. Richter DATE: 04/06/94

APPROVED BY: R.S. GRANTHAM OPERATOR TRAINING SUPERINTENDENT DATE: 04/14/94

Performance Tests

The following performance tests will be conducted (in addition to the annual operability tests) in the second year following certification (Normal Operation Tests, Malfunction Tests, and Transient Tests not tested as Annual Operability Tests):

05-03	Loss of coolant; Outside primary containment
05-06	Failure of safety and relief valves
05-08	Loss or degraded electrical power to the station, including loss of offsite power
05-15	Loss of condenser vacuum including loss of condenser level control
05-16	Loss of service water or cooling to individual components
05-22	Control rod failure including stuck rods, uncoupled rods, drifting rods, rod drops, and misaligned rods
05-23	Inability to drive control rods
05-24	Fuel cladding failure resulting in high activity in reactor coolant or off gas and the associated high radiation alarms
05-27	Failure in automatic control system(s) that affect reactivity and core heat removal
06-08	Plant shutdown from rated power to hot standby
06-09	Cooldown to cold shutdown conditions

PREPARED BY: C. B. Smith DATE: 04/05/94 REVIEWED BY: J. Richter DATE: 04/06/94

APPROVED BY: R.S. GRANTHAM OPERATOR TRAINING SUPERINTENDENT DATE: 04/14/94

Performance Tests

The following performance tests will be conducted (in addition to the annual operability tests) in the third year following certification (Normal Operation Tests, Malfunction Tests, and Transient Tests not tested as Annual Operability Tests):

05-09	Loss or degraded electrical power to the station, loss of emergency power
05-10	Loss or degraded electrical power to the station, loss of emergency generators
05-11	Loss or degraded electrical power to the station, loss of power to the plant's electrical distribution buses
05-31	Main steam line break outside containment
05-33	Main feed line break outside containment
06-01	Plant startup - cold to hot standby. The starting conditions shall be cold shutdown conditions of temperature and pressure. Removal of the reactor vessel head is not a required condition for simulation
06-02	Nuclear startup from hot standby to rated power
06-03	Turbine startup and generator synchronization
06-05	Operations at hot standby
06-06	Load changes
06-10	Core performance testing; Heat Balance

PREPARED BY: C. B. Smith DATE: 04/05/94 REVIEWED BY: J. Richter DATE: 04/06/94

APPROVED BY: R.S. GRANTHAM OPERATOR TRAINING SUPERINTENDENT DATE: 04/14/94

The following performance tests will be conducted (in addition to the annual operability tests) in the fourth year following certification (Normal Operation Tests, Malfunction Tests, and Transient Tests not tested as Annual Operability Tests):

05-12	Loss or degraded electrical power to the station, loss of power to the individual instrumentation buses (AC) that provide power to control room indication or plant control functions affecting the plant's response
05-13	Loss or degraded electrical power to the station, loss of power to the individual instrumentation buses (DC) that provide power to control room indication or plant control functions affecting the plant's response
05-17	Loss of shutdown cooling
05-18	Loss of component cooling system or cooling to individual components
05-19	Loss of normal feedwater or normal feedwater system failure
05-20	Loss of all feedwater (normal and emergency)
05-35	Process instrumentation, alarms, and control system failures
05-36	Passive malfunctions in systems, such as engineered safety features, emergency feedwater systems
05-37	Failure of the automatic reactor trip system
06-14	Operator conducted surveillance testing on safety-related equipment or systems.

A1.1 General

(1) Owner: Georgia Power Corporation

(2) Reference Plant / Type / Rating

(a) Plant E. I. Hatch Unit 1, Doc. No. 50-321
Unit 2, Doc. No. 50-366

(b) General Electric - BWR

(c) Unit 1, 810 MWE
Unit 2, 822 MWE

(3) Date Available for Training: 20 September 1982

(4) Date of Report: May 31, 1990

A1.2 Control Room

NOTE
The simulator is modeled after the Unit Two Reference Plant. The Unit One Reference Plant is very similair to the Unit Two Reference Plant and the Simulator. The documents associated with this report include evaluations on Unit 1 as well as Unit 2 differences.

(1) Control Room Physical Arrangement

The physical layout of the simulator control room is shown in the drawings included with the Evaluation of the Physical Configuration File.

For a listing of differences see Worksheet(s) 04-02, "Differences Between Control Room and Simulator", provided in the Evaluation of the Physical Configuration File.

(2) Panels / Equipment

All Control Room front and back panels necessary to perform operator training are provided, except as noted on Worksheet(s) 04-02.

For a listing of differences see Worksheet(s) 04-02, "Differences Between Control Room and Simulator", provided in the Evaluation of the Physical Configuration File.

(3) Systems

The reference plant systems which are either fully or partially simulated are identified on Worksheet(s) 15-06, "Plant vs Simulator Systems Cross-reference".

(4) Simulator Control Room Environment

The simulator control room duplicates the environment in the Plant Reference Control Room including, communications, lighting, furniture, available computers, and sounds. For a listing of differences see Worksheet(s) 04-02, "Differences Between Control Room and Simulator", provided in the Evaluation of the Physical Configuration File.

A1.3 Instructor Interface

(1) Initial Conditions

The simulator utilizes 999 initial condition sets which are selectable by the instructor. Initial Conditions 101 through 140 are "protected" initial conditions specified by the simulator guides utilized in the operator training programs. They represent a variety of plant states from cold shutdown through power operations and various fuel burnups. Initial Conditions 201 through 899 may be utilized by the instructors to record a specific evolution or event. For a listing of Initial Conditions in which the simulator is certified see Worksheets 13-01, "Listing of Simulator Initial Conditions," provided in the Instructor Controls Capabilities Evaluation File.

(2) Malfunctions

The simulator currently has 265 separate malfunctions. For a listing of Malfunctions in which the simulator is certified see the following documents provided in the Functional Fidelity Evaluation File:

- o Worksheet 05-02, Malfunction Performance Test Abstract.
- o Worksheet 06-02, Normal and Steady State Performance Test Abstract.
- o Worksheet 07-02, Transient Performance Test Abstract.
- o Worksheet 08-02, Simulator Real Time and Operating Limits Abstract.

(3) Remote Functions

For a listing of Remote Functions in which the simulator is certified see Worksheet 13-03, "List of Remote Functions", provided in the Instructor Control Capabilities Evaluation File.

(4) Additional Instructor Station Features Available

For a listing of Instructor Station Features in which the simulator is certified see Worksheet 13-02, "List of Control Functions", provided in the Instructor Control Capabilities Evaluation File. The document titled Simulator Instructor Handbook provides a description of each feature.

A1.4 Operating Procedures for the Reference Plant.

The procedures used during simulator training are controlled Unit 2 documents. Differences between Unit 1 and Unit 2 procedures are documented on Worksheet(s) 13-05, "Unit 1 / Unit 2 Procedure Differences" in the Instructor Control Capabilities Evaluation File.

A1.5 For changes since last report dated May 31, 1990, see attached report.

List of changes since last report

Initial Conditions:

1. We upgraded our Reactor Vessel and Recirc models during this report period. In doing so we have increased the number of Initial Conditions (ICs) that are available. The original ICs are the same except the specific numbers have changed and the total number of ICs has increased to 999. All original ICs numbers have one hundred added to their original number. For a listing of simulator Initial Conditions see Worksheets 13-01, " Listing of Simulator Initial Conditions".

Test Plan changes:

1. During the third year of testing we performed the fourth year tests in order to facilitate simulator model upgrades and testing. Therefore, in the fourth year we performed the third year tests. During a telephone conversation, Mr. Frank Collins stated that this would be acceptable as long as we documented this fact in this section of changes since last report.

Test Changes:

1. A plant design change (# 90-163) deleted the malfunction used in Test 0535 part 3 (Process Instrumentation, Alarms, and Control System Failures, Malfunction # 49 Recirculation Pump Master Flow Controller Failure - High).
2. All test where changed to reflect the new IC numbers described under Initial Conditions of this attachment.

NRC FORM 474
(10-88)
10 CFR § 55.45(b), 55.4 and 55.5

U.S. NUCLEAR REGULATORY COMMISSION
Approved by OMB
3150-0138
Expires 10-31-89

SIMULATION FACILITY CERTIFICATION

INSTRUCTIONS. This form is to be filed for initial certification, recertification (if required), and for any change to a simulation facility performance testing plan made after initial submittal of such a plan. Provide the following information, and check the appropriate box to indicate reason for submittal.

FACILITY
E. I. HATCH, Unit 1, Nuclear Power Station

DOCKET NUMBER
50-321

LICENSEE
Georgia Power Company

DATE
4-19-94

This is to certify that: 1. the above named facility licensee is using a simulation facility consisting solely of a plant-referenced simulator that meets the requirements of 10 CFR § 55.45; 2. the simulation facility meets the guidance contained in ANSI/ANS 3.5, 1985, as endorsed by NRC Regulatory Guide 1.149; and 3. documentation is available for NRC review in accordance with 10 CFR § 55.45(b). If there are any exceptions to the certification of item 2 above, check here ☑ and describe fully on additional pages as necessary.

NAME (or other identification) AND LOCATION OF SIMULATION FACILITY
E. I. Hatch Simulator, US 1 North, Approx. 12 miles North of Baxley, Ga. Zip 31513

XX SIMULATION FACILITY PERFORMANCE TEST ABSTRACTS ATTACHED. (For performance tests conducted in the period ending with the date of this certification)

DESCRIPTION OF PERFORMANCE TESTING COMPLETED (Attach additional page(s) as necessary, and identify the item description being continued)

The Plant E. I. Hatch Unit Two Training Simulator is utilized to perform Unit One Simulator Training. The performance tests were run per the Unit Two Certification. Operational characteristics of Unit One were considered and differences which impact training are identified as exceptions in the Attachment 1 to this form.

X SIMULATION FACILITY PERFORMANCE TESTING SCHEDULE ATTACHED. (For the conduct of approximately 25% of performance tests per year for the four year period commencing with the date of this certification.)

DESCRIPTION OF PERFORMANCE TESTING TO BE CONDUCTED (Attach additional page(s) as necessary, and identify the item description being continued)

The Plant E. I. Hatch Unit Two Training Simulator is utilized to perform Unit One Simulator Training. The performance tests will be run per the Unit Two Certification. Operational characteristics of Unit One will be considered a[illegible] differenc[illegible]s which impact training will be identified.

X PERFORMANCE TESTING PLAN CHANGE (For any modification to a performance testing plan submitted on a previous certification)

DESCRIPTION OF PERFORMANCE TESTING PLAN CHANGE (Attach additional page(s) as necessary, and identify the item description being continued)

See Unit Two SIMULATOR FACILITY CERTIFICATION Form 474.

RECERTIFICATION (Describe corrective actions taken, attach results of completed performance testing in accordance with 10 CFR § 55.45(b)(5)(v). Attach additional page(s) as necessary, and identify the item description being continued.)

Any false statement or omission in this document, including attachments, may be subject to civil and criminal sanctions. I certify under penalty of perjury that the information in this document and attachments is true and correct.

SIGNATURE - AUTHORIZED REPRESENTATIVE	TITLE	DATE
[signature]	Nuclear Plant General Manager	4-19-94

In accordance with 10 CFR § 55.5, Communications, this form shall be submitted to the NRC as follows:

BY MAIL ADDRESSED TO: Director, Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

BY DELIVERY IN PERSON TO THE NRC OFFICE AT: 7920 Norfolk Avenue
Bethesda, MD

Attachment 1

EXCEPTIONS

04/14/94

Plant E. I. Hatch, Unit 1

Docket # 50-321

Georgia Power Company

Facility design and systems relevant to control room personnel, Technical Specifications, Procedures (primarily abnormal and emergency operating procedures), Control room design and instrumentation / control location, and Operational characteristics are generally the same between Unit One and the simulator with exceptions as noted in this attachment. Enclosed in this attachment is a summary of the analysis of the differences between Unit One and the simulator facility in compliance with Regulatory Guide 1.149.

Exceptions to ANSI/ANS 3.5 - 1985 for Unit One are the same as those specified for Unit Two. In addition, Exceptions #010 and #011 are Unit One Operational Characteristics exceptions.

PERFORMED BY: Shaffer DATE: 11/09/89 REVIEWED BY: J. Richter DATE: 04/14/94

APPROVED BY: R. S. Grantham OPERATIONS TRAINING SUPERVISOR DATE: 04/18/94

EXCEPTION NUMBER: 010

ANSI SPECIFICATION: 4.1(3) TITLE: Performance Criteria - Steady State Operation

EXCEPTION: Unit One only

Appendix B of the Standard requires feedwater temperature to be a recorded parameter for steady state performance of a minimum of three points over the power range. Section 3.1 of the Standard requires that the operator shall not observe a difference between the response of the simulator control room instrumentation and the the reference plant. Section 4.1 (3) requires the simulator values for critical parameters shall agree within 2% of the reference plant and shall not detract from training. GPC takes exception to these requirements only to the extent of the feedwater temperature parameter tolerance of within 2% of the Unit One reference plant.

JUSTIFICATION:

Unit One utilizes only the fifth stage of extraction steam from the high pressure turbine for feedwater heating. Unit Two utilizes the fourth and the sixth stages of extraction steam from the high pressure turbine for feedwater heating. The Plant E.I. Hatch Simulator is modeled after the Unit Two reference plant. This results in the feedwater temperature being lower on Unit One than on Unit Two (and therefore the simulator) by more than the + 2% criteria. This difference in feedwater heater and extraction drain system is taught to all licensed personnel in the Initial License Training Program at Plant Hatch.

PERFORMED BY: Rutan & Shaffer DATE: 11/27/89 REVIEWED BY: J. Richter DATE: 04/14/94

APPROVED BY: R. S. Grantham OPERATIONS TRAINING SUPERVISOR DATE: 04/18/94

EXCEPTION NUMBER: 011

ANSI SPECIFICATION: 3.1.2(9) TITLE: Loss of Normal Feedwater or Normal Feedwater System Failure

EXCEPTION: Unit One only

Section 3.1 of the Standard requires that the operator shall not observe a difference between the response of the simulator control room instrumentation and the the reference plant. GPC takes exception to these requirements only for the Unit One feedwater heater extraction pressure indication in the control room on an isolation of the extraction steam to a feedwater heater. Unit One 5th stage heater extraction pressure would decrease vice increase due to the location of the pressure detector; i.e., Unit One pressure is measured downstream of the heater isolation valve - Unit Two pressure is located upstream of the heater isolation valve.

JUSTIFICATION:

On an isolation of extraction steam to a Unit One feedwater heater, extraction pressure and feedwater temperature would decrease. Reactor power would increase in response to the feedwater temperature decrease. The operator would take initial action on the feedwater temperature indication and not on the heater pressure change. The operator would utilize the heater pressure later to diagnose the cause of the feedwater temperature change. On an isolation of extraction steam to a feedwater heater in the simulator, feedwater temperature decreases but extraction pressure increases (simulating Unit Two's response). This difference in extraction pressure indication is taught to all licensed personnel in the Initial License Training Program at Plant Hatch.

Plant E. I. Hatch, Unit 1 Docket # 50-321 Georgia Power Company

Attachment 1 04/14/94

Hardware and Environmental Differences

The E.I. Hatch simulator is modeled after the Unit Two control room. The specific hardware and environmental differences between Unit One and Unit Two Control Rooms were assembled and evaluated to determine the significance of each (major differences which could have a negative impact if a component, component location or environmental condition were not identical). Any noted differences have been evaluated for there potential training impact and are addressed in the Operator Initial License Training Program and in the Licensed Operator Requalification Training Program, as applicable.

The panels that have been simulated in the E.I. Hatch simulator maintain correct physical relationships to the panels of the Unit One and Unit Two control rooms with minor exceptions related to Unit One back panel locations. Control room design and instrumentation/control location are the same between Unit One and Unit Two except as noted below. Environmental conditions (noise level, AC and DC lighting levels, etc.) were evaluated with only minor differences existing.

A summary of the areas identified as differences are as follows:

- The simulator includes the control panels for the MSIV Leakage Control System and Hydrogen Recombiners associated with Unit Two. These panels are not located in the Unit One control room.
- The simulator models the Drywell Chillers and Drywell Cooler arrangement of Unit Two. Unit One uses a similar Drywell Cooler arrangement but uses Plant Service Water as the cooling medium vice using Drywell Chillers.
- Unit One uses a fifth stage Feedwater Heater with the associated control, indications, and annunciators on front panel P650. Unit Two uses a fourth and sixth stage Feedwater Heater with the associated control and indications on front panel P650 with the annunciators located on back panel P656 (operators are prompted on front panel P650 by a Feedwater Heater Trouble annunciator).
- The Unit One control panels contain controls and annunciators for the Main Stack Dilution system, Recombiner Building Ventilation system, and the Main Control Room Environmental Control system. Unit Two does not control these plant systems.
- The controls and mimic for the LPCI inverters and buses (S018A and S081B) are located on front panel P601 for Unit One and on front panel P652 for Unit Two.
- Nomenclature and MPL label plate wording is not always identical. For example, the label for a Unit One relief valve reads "LLS/Manual Relief Vlv 1B21-F013H". The label for the corresponding relief valve for Unit Two "Auto Relief 2B21-F013H".
- The Control Building Ventilation system is controlled from Unit One back panels. Unit Two does not control this plant system.

GPC feels the differences noted during the in-depth hardware and environmental review to be acceptable with respect to use of the E.I. Hatch simulator for operator training for both Unit One and Unit Two.

Operating Procedure Differences

The unit specific operating procedures for both units were compared side by side on a one for one basis using the Unit Two operating procedures as the comparison measure. The Unit Two procedures are used in the simulator and require no modification or changes from those used in the plant. The specific differences between the Unit One and Unit Two procedures were assembled and compared to determine the significance of each (major differences which could have a negative effect if using Unit Two operating procedures for operation of Unit One) and to determine the disposition to address these differences in the E.I. Hatch Licensed Operator Training programs.

The comparison of Unit One and Unit Two procedures focused on the procedures identified for use in simulator training scenarios as part of the Licensed Operator Training Program Master Plans. Procedures not used for simulator training and not specified by the Master Plan were not addressed in this comparison. The types of procedures addressed in this comparison included System, Abnormal, and Emergency Operating procedures, Surveillance Procedures, and Administrative Control procedures.

The Unit One and Unit Two operating procedures are essentially identical. Minor differences concerning specific equipment and operating practices were noted during the evaluation. Differences that have been identified between each unit's procedures are addressed in the Operator Initial License Training Program and in the Licensed Operator Requalification Training Program, as applicable.

A summary of the areas identified as differences are as follows:

- Unit specific components are covered only by the procedures associated with that unit. (Unit Two has procedures covering the operation of the Post LOCA Hydrogen Recombiner system, the Standby Plant Service Water pump for the B Diesel Generator, and the Drywell Chillers. Unit One does not have these systems.)

- Unit specific surveillance procedures are used to address surveillance items not required by both units (e.g., "MSIV EXERCISE" testing is not required for Unit 2).

- Emergency Operating Procedures provide specific MPL numbers for those indications required to be monitored for an action step in the flow charts, abnormal, and EOP 100 series procedures. In some cases, the MPL number for a specific parameter is different between Unit One and Unit Two. (e.g. Different MPL numbers are specified for an RTD at the same location in both units when used to determine average drywell temperature.)

- Differences in setpoints, entry conditions, and various other operational differences have also been addressed.

GPC feels these differences are acceptable with respect to the use of the E.I. Hatch simulator for operator training for both Unit One and Unit Two.

Technical Specification Differences

The Technical Specifications for Unit One and Unit Two were compared side by side on a one for one basis using the Unit Two Technical Specifications as the comparison measure. The specific differences between the two units were assembled and compared to determine the significance of each (major differences which could have a negative effect when using Unit Two Technical Specifications for operation of Unit One) and to determine the disposition to address these differences in the E.I. Hatch Licensed Operator Training programs.

The Unit One and Unit Two Technical Specifications are very similar with regard to the items covered and the approach to inoperable equipment. Differences that have been identified between each unit's specifications are addressed in the Operator Initial License Training Program and in the Licensed Operator Requalification Training Program, as applicable.

A summary of the areas identified as differences are as follows:

- Unit One Technical Specifications are written in a two (vertical) column format. Unit Two Technical Specifications are written in a top to bottom (horizontal) format.

- Inoperable components covered by Technica Specifications require similar actions for both units. Some differences associated with the Limiting Conditions for Operation involve the level of shutdown required when an LCO cannot be met (Unit One typically requires Cold Shutdown, Unit Two typically requires Hot Shutdown) and the specified time requirements to meet the shutdown ACTION statements of an LCO (Unit One typically allows 24 hours, Unit Two typically allows 12 hours). These differences are acceptable because the LCO ACTION requirement is a shutdown in both cases. The type of shutdown and the time requirements would be addressed when writing the LCO status sheet, which is required to written well before either time requirement would expire.

- Specification numbers for specific components are different between the units. (e.g. Unit One covers A.C. Power Sources in section 3.9, Unit Two covers A.C. Power Sources in section 3.8)

- Unit specific components are covered only by the Technical Specifications associated with that unit. (i.e. Unit One has specification regarding the Recombiner Building components. Unit Two does not have a Recombiner Building. Unit Two has a Hydrogen Recombiner system covered by Technical Specifications. Unit One does not have thissystem.)

- Surveillances intervals differ in many instances for a specific component on Unit One or Unit Two. These differences are administratively controlled in the Test and Surveillance Control procedure, 90AC-OAP-001-0S. This procedure is used plant wide and addresses surveillance requirements for both units by the generation of a surveillance requirements schedule. Additional administrative controls are incorporated into plant wide procedures to address these differences (e.g. Thermal Limits are required to be monitored more frequently per Unit Two Technical Specifications during power level changes and when operating under certain limiting conditions. Plant procedure, 34SV-SUV-020-0S, requires this increased surveillance to be performed for either unit when operating under these conditions.)

Continued

- Differences in allowable run times with Inoperable components, minor Technical Specification setpoint differences and differences in the minimum number of operable components were also addressed.

GPC feels these differences are acceptable with respect to the use of the E.I. Hatch simulator for operator training for both Unit One and Unit Two.

Plant E. I. Hatch is in the development stage of piloting the new Technical Specification Improvement Program that will aid in minimizing the differences existing at present.

C

WOLF CREEK
NUCLEAR OPERATING CORPORATION

Neil S. "Buzz" Carns
President and
Chief Executive Officer

May 20, 1994

WM 94-0081

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Reference: Letter ET 91-0235, dated December 24, 1991, from F. T. Rhodes, WCNOC, to NRC

Subject: Docket No. 50-482: Revision to NRC Commitment Made in WCNOC's Response to Generic Letter 88-20, Supplement 4

Gentlemen:

The purpose of this letter is to advise the NRC of a revision to a commitment made in the Reference. The Reference provided Wolf Creek Nuclear Operating Corporation's (WCNOC) response to Generic Letter 88-20, Supplement 4: "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)." In response to the generic letter, WCNOC committed to perform a seismic level 1 Probabilistic Risk Assessment(PRA) plus a containment performance analysis, a fire PRA, and the use of a screening type approach shown in Figure 1 of Generic Letter 88-20, Supplement 4 to evaluate the impact of high winds, external floods, and transportation and nearby facility accidents. This commitment has been revised to perform a reduced scope seismic evaluation in lieu of a seismic level 1 PRA plus a containment performance analysis.

Generic Letter 88-20, Supplement 4, and NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," provided a basis for each plant to examine seismic events. A methodology for establishing a Review Level Earthquake was presented, along with a binning procedure to group plants according to seismic hazard levels as Reduced Scope plants, 0.3g Focused Scope plants, 0.3g Full Scope plants, and plants committed to perform a seismic PRA. Wolf Creek Generating Station (WCGS) was placed in the Focused Scope bin. This required a seismic level 1 PRA plus a containment performance analysis or a seismic margin methodology. Nuclear plants placed in the Reduced Scope bin were only required to perform detailed plant walkdowns.

Recently NUREG-1488, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains," was issued. Revisions of the Lawrence Livermore National Laboratory (LLNL) seismic hazard methodology over the past fifteen years have resulted in a continuous decrease in the assessed annual probability of exceeding the Safe Shutdown Earthquake (SSE) for all Eastern United States plants. These revised LLNL hazard results corroborate those previously developed by the Electric Power Research

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Institute (EPRI) and have confirmed that the seismic hazard at most Eastern United States plants is low, comparable to the 1989 LLNL seismic hazard results at 10 sites which were binned as reduced scope plants in NUREG-1407. This information along with quantitative core damage values based on conservative plant Safe Shutdown Earthquake capacities support the position that Focused Scope plants (WCGS) should perform a Reduced Scope seismic review.

In 1993, recognizing the value of detailed plant walkdowns as the most important tool for identifying seismic weak links, preparations for walkdowns began at WCGS. The preparatory work has included all of the rigor necessary to assure that front line systems and components, support systems and components, and plant unique features are reviewed during the walkdown. In March 1994, also as part of the preparatory work for the walkdowns and for use in the later analysis and development of component fragilities, WCGS began to select the seismic hazard estimates, establish the seismic spectral shape, and develop the in-structure seismic demand. During this activity, EPRI NP-6395-D, NUREG/CR-5250, and NUREG-1488 were reviewed in detail. On the basis of this review, and after discussions with other utilities and industry groups, WCGS concluded that the goals of the seismic IPEEE program could be met with a Reduced Scope program as defined in NUREG-1407. Therefore, WCGS will complete the seismic portion of the IPEEE using an approach and methodology consistent with the Reduced Scope Margins Method provided in NUREG-1407. This approach will emphasize well conducted, detailed walkdowns as the tool for identifying potential seismic weak links. As discussed in NRC Information Notice 94-32, "Revised Seismic Hazard Estimates," the NRC staff is reviewing the information in NUREG-1488 to assess if it is appropriate to revise the IPEEE seismic scope. WCNOC has reviewed this information and has determined that a Reduced Scope Margins Method is appropriate for WCGS and is proceeding with this method. The results of the WCGS IPEEE will be submitted by June 30, 1995, as currently committed. WCNOC requests that prompt notification be made if the NRC determines that it is not appropriate to revised the IPEEE seismic scope due to the impact on resources needed to meet the June 30, 1995 schedule.

If you have any questions concerning this matter, please contact me at (316) 364-8831, extension 4000, or Mr. Kevin J. Moles at extension 4565.

Very truly yours,

Neil S. Carns

Neil S. Carns

NSC/jra

cc: L. J. Callan (NRC)
G. A. Pick (NRC)
W. D. Reckley (NRC)
T. Reis (NRC)

CP&L

Carolina Power & Light Company
Robinson Nuclear Plant
PO Box 790
Hartsville SC 29550

Robinson File No.: 13510E
Serial: RNP/94-1033

MAY 1 9 1994

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
COMPLETION DATE FOR REACTOR AUXILIARY BUILDING VENTILATION SYSTEM FLOW BALANCING

Gentlemen:

The purpose of this letter is to inform you of a change in the date for the completion of flow balancing of the Reactor Auxiliary Building Ventilation system at the H. B. Robinson Steam Electric Plant, Unit No. 2. Carolina Power & Light Company had committed to complete this work by May 20, 1994, as delineated in our letter dated October 11, 1993, (Serial: RNP/93-2535). However, due to emergent work involving the Control Room Ventilation system, it has become necessary to reschedule the completion date until June 15, 1994.

Questions regarding this matter may be referred to Mr. K. R. Jury at (803) 383-[illegible]63.

Very truly yours,

RM Krich

R. M. Krich
Manager - Regulatory Affairs

c: Mr. S. D. Ebneter, Regional Administrator, USNRC Region II
Ms. B. L. Mozafari, USNRC Project Manager, HBRSEP
Mr. W. T. Orders, USNRC Senior Resident Inspector, HBRSEP

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Consolidated Controls™
Products

LTR1858

EATON

May 10, 1994

United States Nuclear Regulatory Commission
Document Control Desk (Part 21)
Washington, DC 20555-0001

Reference: Northeast Utilities Letter MP2-I-94-040
Eaton Letter LTR1856

Subject: 10CFR21 Reporting for Failures of 6N642 Undervoltage Modules

Gentlemen:

As was reported in Eaton Letter LTR1856, Eaton agrees with Northeast Utilities' assessment that the design of the 6N642 Module causes a higher than normal failure rate of an integrated circuit (IC) on that electronic module assembly. The purpose of this letter is to provide additional information to Northeast Utilities, inform the NRC of the details of the situation, fulfill the reporting requirements of 10CFR21, and to provide a plan for resolving this issue.

A brief history is necessary to fully understand this issue. This electrical module is part of the Engineered Safeguards Actuation System (ESAS) at the Millstone II site. Failures have occurred with an IC (part U7) on this module. In July of 1993, the schematic for this module S6N642 was modified (to Revision F) to improve the circuit design associated with U7 and submitted to Northeast Utilities. The associated artwork for the printed circuit board (PCB, part number KTB7361) was upgraded (to Revision D) in November 1993 to incorporate the change as shown on the modified schematic.

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FAX 203 / 796-6313
TWX 310 / 681-9305
Telex 681-9305

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In September of 1993 (before the artwork Revision D release date) an order was placed by NU for spare modules (6N642) and parts were procured by Eaton to Revision C of the artwork which was active at that time. When Revision D of the artwork was released in November 1993, the spare order was not updated due to mis-communication about the Engineering Change Notice. Subsequently it was determined that the spare order needed to be updated and it was determined that the modules in the field (existing spares and units installed in ESAS) should also be updated in order to improve the reliability of this assembly.

Eaton has concluded the investigation of the events described above and concurs with NU's findings relative to the timing of these events. Upon further review of the schematic by an independent engineer, it was determined that an additional change to the circuit associated with U7 is advisable in order to reduce the likelihood that further reliability issues will surface. It was determined that two spare inputs (U7 pins 11 and 12) were left unconnected and floating. The manufacturers for these CMOS devices (U7) recommend pulling all spare inputs to ground. Failure to do so could cause a higher power dissipation (as with the previous change) and therefore a higher operating temperature. A higher operating temperature is usually associated with lower reliability (temperature and reliability are usually inversely related).

To finally resolve this issue, Eaton will take the following course of action:

1) The schematic and artwork will be upgraded again, (to Revisions G and E, respectively) to incorporate this latest change within two weeks from the date of this letter.

2) The spare order presently in house will be upgraded again (the previous change had been completed), to incorporate this latest change. The witness of testing and auditing for this spare order by Northeast Utilities should be postponed until this new change is incorporated (upgrade to be completed by June 3, 1994).

3) Eaton recommends that NU return the two existing spare units for upgrade by Eaton, after the present spare order for four units is shipped.

4) Eaton recommends that NU swap extra spares for operational units (maintaining two updated spares at all times) and return of the operational units (eight total) to Eaton for upgrading.

5) Eaton will write and release a Field Change Procedure so that the units can be upgraded on site if necessary (it is preferable to upgrade the units at Eaton's facility if at all possible).

6) Eaton will update these units during the next outage (scheduled for the beginning of August 1994) if necessary.

Eaton will work with NU to resolve this issue as soon as possible in order to minimize any inconvenience. We invite further comment on the action plan outlined above.

Sincerely;

G DeRome

Gerard DeRome
Manager Power Industry Controls

Copy:

Attn: Walter Haass
Vendor Inspection Branch
United States Nuclear Regulatory Commission
Washington DC 20555-0001

Eaton Corporation
Pressure Sensors Division
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Bethel, Connecticut 06801
203 / 796-6000

Consolidated Controls™ Products

LTR 1856

May 6, 1994

Daniel Lowell
I&C Department
Millstone Unit II
P.O. Box 128
Waterford, CT 06358-0128

Subject: Failures of 6N642 Undervoltage Modules

Reference: NU Letter MP2-I-94-040

Mr. Lowell

Eaton Consolidated Controls appreciates your notification of the suspected problem with the design of the 6N642 Module. Eaton has initiated an effort to investigate this issue and determine the cause of the problem and define a solution that best fits the needs of Northeast Utilities and Eaton. NU's assessment of the problem appears correct and confirmation should be completed early next week along with a proposed solution.

As far as the 10-CFR-21 reporting requirements are concerned, NU is obviously aware of the problem with this product and Eaton is confirming, via this letter, that there appears to be a design defect in the modules shipped that was not corrected, because the design change was not incorporated into that shipment of modules. The impact of this problem is limited to the equipment installed in Millstone Unit II.

Eaton will provide further response on this issue by May 13, 1994.

Sincerely,

Gerard DeRome
Manager Power Industry Controls

NORTHEAST UTILITIES
THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

MEMO

May 6, 1994 MP2-I-94-040

Gerry DeRome
Manager-Power Industry Controls
Eaton Corporation
9 South Street
Danbury, CT 06810

Daniel Lowell
I&C Department
Millstone Unit II
P.O. Box 128
Waterford, CT 06385-0128

Subject: Failures of 6N642 Undervoltage modules.

Mr. DeRome,

As you are aware, the Millstone II Engineered Safeguards Actuation System has experienced numerous failures of it's Undervoltage Input modules, Eaton Part number 6N642. These failures have occurred over the last 12 months, and have resulted in six out of the eight installed modules failing. Each failure has been associated with the same component, analog switch U7.

After a review of the module schematics, it appears that an effort to correct this design shortcoming was initiated during July of 1993. This modification moved the voltage source for R22 and R25 from +5 VDC to +12 VDC, Reference S6N642; Rev. F. This should correct the problem of U7 failing prematurely by bringing the control and supply voltages to the same level.

An order to purchase four additional spare 6N642 modules, Northeast Utilities Purchase Order Number 287353, was placed on September 17, 1993. These new modules do not contain the latest revision of the design. The Printed Wiring Boards used in these modules are Revision C, and should be Revision D. The modules must be modified to conform to Revision F of S6N642.

The eight modules currently installed in our ESAS and the two spare modules in our warehouse must be modified to correct the design flaw. A field change procedure, or similar document, must be created to modify these ten modules. This procedure is required as soon as possible, as new module failures continue to occur on a regular basis. The potential for multiple channels of ESAS failing in the non-conservative direction makes the swift completion of these repairs very important to Northeast Utilities.

A workable program to revise these modules will need to be developed. Depending on the outcome of discussions with the various organizations involved, two possible options exist: The first method being that a representative of your organization comes to our facility to perform the modifications. The modules would then be retested using normal plant procedures to verify that they are operable. The second option is that when the additional spare modules have been delivered to us, we send the older revision modules back in small groups to be modified. Groups of three or four modules can be return to your factory to be revised, until all ten older revision modules have been updated. A fairly quick turn-around, approximately one week, will be required on the returned modules.

After reviewing 10-CFR-21, and discussing its intent with other members of the Millstone II staff, we have concluded that this defect is reportable under 10-CFR-21. Eaton Corporation should take the necessary steps to initiate this process as soon as possible.

If you have any questions on this or any other issue concerning ESAS please feel free to contact me at (203) 447-1791 X6740. Thank you for your prompt attention to this issue.

Daniel F. Lowell
Millstone Unit II
I&C Department

Copy: M. Bain
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T. Arnett
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Northeast
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Hartford, CT 06141-0270
(203) 665-5000

May 18, 1994

Docket No. 50-423
B14839

Re: 10CFR50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Nuclear Power Station, Unit No. 3
Proposed Revision to Technical Specifications
Fuel Building Exhaust Filter System

Pursuant to 10CFR50.90, Northeast Nuclear Energy Company (NNECO) hereby proposes to amend its Operating License NPF-49, by incorporating the change identified in the attachments into the Technical Specifications of Millstone Unit No. 3.

Description of Proposed Change

The proposed change to the Technical Specifications of Millstone Unit No. 3 is being made to better define those periods of time during which the Fuel Building Exhaust Filter System has been shown to be required. The Millstone Unit No. 3 Final Safety Analysis Report (FSAR), Section 9.4.2.2, states that "fuel building filtration is required whenever heavy loads (fuel or other) are moved within or over the spent fuel pool and less than 60-day [decay time] fuel is in the pool." Operation of the filter system during other periods of time is considered unnecessary and will decrease the charcoal adsorption efficiency.

The proposed change to Technical Specification 3/4.9.12, "Fuel Building Exhaust Filter System," will result in a modification to the Applicability, Surveillance Requirement, and Bases sections.

The Applicability section of Technical Specification 3/4.9.12 is proposed to be modified to indicate that the Fuel Building Exhaust Filter System is required to be operable whenever irradiated fuel is in the spent fuel pool, which has had less than 60 days of decay time. Per the offsite dose analysis performed to support this proposed change, operation of the ventilation system with fuel that has decayed for greater than 60 days is not required.

Surveillance Requirement 4.9.12a is proposed to be modified to require that the system be tested and verified operable at no

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greater than 31 days prior to its required usage. The Fuel Building Exhaust Filter System is required to be operable whenever irradiated fuel, which has decayed less than 60 days, is in the pool.

Bases section 3/4.9.12, "Fuel Building Exhaust Filter System," is proposed to be modified to reflect the key assumptions and results of the offsite dose calculation which supports FSAR Section 9.4.2.2 and this proposed change.

Safety Assessment

The proposed change will modify the operability requirements for the Fuel Building Exhaust Filter System. Currently, the system is required to be operable whenever fuel is moved in the storage area and whenever there is a load transported over the area. In addition, the system's Surveillance Requirement requires that the system must be verified operable once every 31 days. The proposed modification will require the system to be operable whenever irradiated fuel, which has decayed less than 60 days, is in the spent fuel pool and will require that the system be verified operable within 31 days of load movement or fuel movement over the pool which contains fuel which has decayed less than 60 days.

FSAR Section 15.7.4, "Design Basis Fuel Handling Accidents" discusses the effects of dropping a spent fuel assembly on to another fuel assembly in the spent fuel pool. This event assumes the rupturing of the cladding of all the fuel rods in the dropped assembly and fifty fuel rods in the second assembly. This accident analysis has been reviewed and is not negatively affected by this proposed change.

The most critical time, with respect to offsite dose consequences, for a fuel handling accident is immediately after a reactor shutdown since radioiodine concentrations are the greatest. However, Technical Specification 3/4.9.3 restricts fuel movement until the fuel has decayed for at least 100 hours. Therefore, 100 hours is the most critical time in this fuel handling accident. In this event, with the proposed technical specification, the filtration system is required to be operable. Therefore, the resultant offsite doses are identical to those contained in the FSAR. The other critical time is when the fuel has decayed for sixty days and the Fuel Building Exhaust Filter System is isolated. A fuel handling accident at this period of time would result in an unfiltered release to the atmosphere. A calculation was performed at t=60 days using the same assumptions as were used in the calculation for t=100 hours, except that no credit was taken for filtration and the inventory available for release was decreased to correspond to the additional decay time. The results of the offsite dose calculation for t=60 days shows that the dose to the thyroid is 1.5 REM and dose to the whole body is 0.02 REM. The fuel handling accident discussed in the FSAR shows that the thyroid dose at t=100 hours is 7.6 REM and the whole body dose at t=100 hours is 0.51 REM. As can be seen, the existing accident analysis is more limiting. Therefore, the isolation of the Fuel Building Exhaust Filter System after irradiated fuel has decayed for 60 days or more is safe.

The two other fuel handling accidents, namely a fuel drop in containment and a cask drop are unaffected by this change. The fuel drop in containment is unaffected since the Fuel Building Exhaust Filter System does not serve containment. The consequences of a cask drop does not need to be re-evaluated since the potential cask drop distances are less than 30 feet and the appropriate impact limiting devices are employed during cask movements. Therefore, a spent fuel cask drop accident is not limiting and is less severe than a fuel handling accident and has not been re-evaluated. This position is consistent with the Millstone Unit No. 3 Final Safety Analysis Report.

Significant Hazards Consideration

In accordance with 10CFR50.92, NNECO has reviewed the attached proposed change and has concluded that it does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(C) are not compromised. The proposed change does not involve an SHC because the change would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed modification will revise the period of time during which the Fuel Building Exhaust Filter System must be operable. The propose change will require that the system is operable whenever irradiated fuel, which has decayed less than 60 days, is in the spent fuel pool. Currently, the system is required to be operable whenever a load is moved over the pool or fuel is being moved in the pool.

   The modification has no effect on the probability of a fuel handling accident. The consequences of a fuel handling accident has been evaluated at two intervals. The first time is the minimum decay time. At this time (t=100 hours) with irradiated fuel in the pool, the Fuel Building Exhaust Filter System is required, per the existing and the proposed Technical Specification, to be operable. Therefore, the consequences of an accident are identical to that described in the FSAR. The second scenario evaluated is when the filters are initially isolated (t=60 days). The resultant offsite dose, assuming no filtration and lower core inventory due to decay, are significantly lower than was calculated at t=100 hours. Therefore, the existing accident analysis in FSAR Section 15.7.4 is limiting and the proposed modification will not impact the probability or consequences of an accident.

2. Create the possibility of a new or different kind of accident from any previously evaluated.

   The proposed change does not impact any system or component which could cause a fuel handling accident. The Fuel Building Exhaust Filter System is used for accident mitigations. It's failure cannot, in any way, create the possibility of a new or different kind of accident.

3. Involve a significant reduction in a margin of safety.

   The proposed change to the Fuel Building Exhaust Filter System has been analyzed at the two most critical times. The first analysis was done when the fuel is first placed in the pool, and the second analysis was done when the filtration system is isolated. The first event resulted in no change in assumptions in the analysis presented in the FSAR, ergo no change in dose. The second event has been analyzed and doses have decreased, when compared to the first event. The system will be verified operable per the performance of Surveillance Requirement 4.9.12a prior to fuel or load movement over the pool. Therefore, there is no reduction in the margin of safety.

Moreover, the Commission has provided guidance concerning the application of standards in 10CFR50.92 by providing certain examples (51FR7751, March 6, 1986) of amendments that are considered not likely to involve an SHC. Although the proposed change is not enveloped by a specific example, the proposed change would not involve a significant increase in the probability or consequences of an accident previously analyzed. The proposed change has been analyzed and it has been shown that the Accident Analysis in FSAR Section 15.7.4 is limiting. The probability of a new or different kind of accident or a reduction in the margin of safety will not occur.

NNECO has reviewed the proposed license amendment against the criteria of 10CFR51.22 for environmental considerations. The proposed change does not involve an SHC, nor increase the types and amounts of effluents that may be released offsite, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, NNECO concludes that the proposed change meets the criteria delineated in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an environmental impact statement.

The mark-up of the existing Technical Specifications is contained in Attachment 1. The retype of the proposed change to the Technical Specifications is contained in Attachment 2 and reflects the currently issued version of Technical Specifications.

The Millstone Unit No. 3 Nuclear Review Board has reviewed and approved this proposed amendment and concurs with the above determination.

In accordance with 10CFR50.91(b), we are providing the State of Connecticut with a copy of this proposed amendment.

Regarding our proposed schedule for this amendment, we request issuance at your earliest convenience with the amendment effective as of the date of issuance, to be implemented within 30 days of issuance.

Should the Staff require any additional information, please contact Mr. R. G. Joshi at (203) 665-3844.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

J. F. Opeka

J. F. Opeka
Executive Vice President

cc: T. T. Martin, Region I Administrator
V. L. Rooney, NRC Project Manager, Millstone Unit No. 3
P. D. Swetland, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3

Mr. Kevin T.A. McCarthy, Director
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
P.O. Box 5066
Hartford, CT 06102-5066

Subscribed and sworn to before me

this 18th day of May, 1994

Lorraine J. D'Amico

Date Commission Expires: 3/31/98

Docket No. 50-423
B14839

Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Proposed Revision to Technical Specifications
Fuel Building Exhaust Filter System

Marked Up Pages

May 1994

3/4.9.12 FUEL BUILDING EXHAUST FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent Fuel Building Exhaust Filter Systems shall be OPERABLE. At least one Fuel Building Exhaust Filter System shall be in operation whenever any evolution involving movement of fuel within the storage pool or crane operations with loads over the storage pool is in progress.

APPLICABILITY: Whenever irradiated fuel with less than 60 days decay is in the storage pool.

ACTION:

a. With one Fuel Building Exhaust Filter System inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE Fuel Building Exhaust Filter System is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no Fuel Building Exhaust Filter System OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one Fuel Building Exhaust Filter System is restored to OPERABLE status.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel Building Exhaust Filter Systems shall be demonstrated OPERABLE:

a. ~~At least once per~~ Within 31 days ~~on a STAGGERED TEST BASIS~~ by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying a system flow rate of 20,700 cfm ±10% and that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

Prior to moving fuel within or loads over the storage pool when irradiated fuel with less than 60 days decay is present.

0227

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 FUEL BUILDING EXHAUST FILTER SYSTEM

The limitations on the Fuel Building Exhaust Filter System ensure that all radioactive ~~material~~ iodine released from an irradiated fuel assembly and storage pool water will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. A

see next sheet

3/4.9.13 SPENT FUEL POOL - REACTIVITY

The limitations described by Figure 3.9-1 ensure that the reactivity of fuel assemblies introduced into Region II are conservatively within the assumptions of the safety analysis.

Administrative controls have been developed and instituted to verify that the enrichment and burn-up limits of Figure 3.9-1 have been maintained for the fuel assembly.

3/4.9.14 SPENT FUEL POOL - STORAGE PATTERN

The limitations of this specification ensure that the reactivity conditions of the Region I storage racks and spent fuel pool k_{eff} will remain less than or equal to 0.95.

The Cell Blocking Devices in the 4th location of the Region I storage racks are designed to prevent inadvertent placement and/or storage of fuel assemblies in the blocked locations. The blocked location remains empty to provide the flux trap to maintain reactivity control for fuel assemblies in adjacent and diagonal locations of the STORAGE PATTERN.

STORAGE PATTERN for the Region I storage racks will be established and expanded from the walls of the spent fuel pool per Figure 3.9-2 to ensure definition and control of the Region I/Region II boundary and minimize the number of boundaries where a fuel misplacement incident can occur.

INSERT A FOR BASIS 3/4.9.12

The filtration system removes radioiodine following a fuel handling or heavy load drop accident. Noble gases would not be removed by the system. Other radionuclides would be scrubbed by the storage pool water. Iodine-131 has the longest half-life: ~8 days. After 60 days decay time, there is essentially negligible iodine and filtration is unnecessary.

Docket No. 50-423
B14839

Attachment 2

Millstone Nuclear Power Station, Unit No. 3

Proposed Revision to Technical Specifications
Fuel Building Exhaust Filter System

Retyped Pages

May 1994

3/4.9.12 FUEL BUILDING EXHAUST FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent Fuel Building Exhaust Filter Systems shall be OPERABLE. At least one Fuel Building Exhaust Filter System shall be in operation whenever any evolution involving movement of fuel within the storage pool or crane operations with loads over the storage pool is in progress.

APPLICABILITY: Whenever irradiated fuel with less than 60 days decay is in the storage pool.

ACTION:

a. With one Fuel Building Exhaust Filter System inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE Fuel Building Exhaust Filter System is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no Fuel Building Exhaust Filter System OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one Fuel Building Exhaust Filter System is restored to OPERABLE status.

c. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel Building Exhaust Filter Systems shall be demonstrated OPERABLE:

a. Within 31 days prior to moving fuel within or loads over the storage pool when irradiated fuel with less than 60 days decay is present by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers, and verifying a system flow rate of 20,700 cfm ±10% and that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 FUEL BUILDING EXHAUST FILTER SYSTEM

The limitations on the Fuel Building Exhaust Filter System ensure that all radioactive iodine released from an irradiated fuel assembly and storage pool water will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. The filtration system removes radioiodine following a fuel handing or heavy load drop accident. Noble gases would not be removed by the system. Other radionuclides would be scrubbed by the storage pool water. Iodine-131 has the longest half-life: ~8 days. After 60 days decay time, there is essentially negligible iodine and filtration is unnecessary.

3/4.9.13 SPENT FUEL POOL - REACTIVITY

The limitations described by Figure 3.9-1 ensure that the reactivity of fuel assemblies introduced into Region II are conservatively within the assumptions of the safety analysis.

Administrative controls have been developed and instituted to verify that the enrichment and burn-up limits of Figure 3.9-1 have been maintained for the fuel assembly.

3/4.9.14 SPENT FUEL POOL - STORAGE PATTERN

The limitations of this specification ensure that the reactivity conditions of the Region I storage racks and spent fuel pool k_{eff} will remain less than or equal to 0.95.

The Cell Blocking Devices in the 4th location of the Region I storage racks are designed to prevent inadvertent placement and/or storage of fuel assemblies in the blocked locations. The blocked location remains empty to provide the flux trap to maintain reactivity control for fuel assemblies in adjacent and diagonal locations of the STORAGE PATTERN.

STORAGE PATTERN for the Region I storage racks will be established and expanded from the walls of the spent fuel pool per Figure 3.9-2 to ensure definition and control of the Region I/Region II boundary and minimize the number of boundaries where a fuel misplacement incident can occur.

Department of Energy
Office of Civilian Radioactive Waste Management
Yucca Mountain Site Characterization Office
P.O. Box 98608
Las Vegas, NV 89193-8608

WBS 9.1.2
QA: N/A

MAY 19 1994

Larry R. Hayes
Technical Project Officer
for Yucca Mountain
Site Characterization Project
U.S. Geological Survey
101 Convention Center Drive
Suite 860
Las Vegas, NV 89109

ISSUANCE OF SURVEILLANCE RECORD YMP-SR-94-042 RESULTING FROM YUCCA MOUNTAIN QUALITY ASSURANCE DIVISION (YMQAD) SURVEILLANCE OF U.S. GEOLOGICAL SURVEY (USGS) (SCP: N/A)

Enclosed is the record of Surveillance YMP-SR-94-042 conducted by the YMQAD at the USGS facilities at Yucca Mountain, Nevada, site March 8 and 15, 1994, and April 21, 1994.

The purpose of the surveillance was to verify the implementation of USGS Technical Procedure NWM-USGS-HP-254 and the consequent development of the calibration equation for neutron moisture meters.

No Corrective Action Requests were issued as a result of this surveillance. This surveillance is considered completed and closed as of the date of this letter. A response to this surveillance record and any documented recommendations is not required.

If you have any questions, please contact either Robert B. Constable at 794-7945 or Raul A. Hinojosa at 794-7991.

Robert B Constable for

Richard E. Spence, Director
Yucca Mountain Quality Assurance Division

YMQAD:RBC-3565

Enclosure:
Surveillance Record
YMP-SR-94-042

Add: Bill Belke Au, Encl

NH03
11
WM-11
102.7

250031

YMP-5

9405250241 940519
PDR WASTE
WM-11 PDR

Surveillance No. YMP-SR-94-042

OFFICE OF
RADIOACTIVE WASTE MANAGEMENT
U.S. DEPARTMENT OF ENERGY
WASHINGTON, D.C.

QUALITY ASSURANCE SURVEILLANCE RECORD

SURVEILLANCE DATA

1 ORGANIZATION/LOCATION:	2 SUBJECT:	3 DATE: 3/8/94
U.S. Geological Survey (USGS) Yucca Mountain Site Area 25	Neutron Moisture Meter Calibration Equation	

4 SURVEILLANCE OBJECTIVE:
To determine that USGS develops the calibration equation for neutron moisture meters.

5 SURVEILLANCE SCOPE:
The test verification of neutron moisture meters and the development of the calibration equations in accordance with USGS procedure HP-254, Revision 0.

6 SURVEILLANCE TEAM:
Team Leader:
Raul A. Hinojosa
Additional Team Members:

7 PREPARED BY: Raul A. Hinojosa
Raul A. Hinojosa, Surveillance Team Leader — Date: 3/4/94

8 CONCURRENCE: Robert B. Constable for
QA Division Director — Date: 3/4/94

SURVEILLANCE RESULTS

9 BASIS OF EVALUATION/DESCRIPTION OF OBSERVATIONS:

See Page(s) 2

10 SURVEILLANCE CONCLUSIONS:

See Page(s) 2

11 COMPLETED BY: Raul A. Hinojosa
Surveillance Team Leader — Date: 5/17/94

12 APPROVED BY: Robert B. Constable for
QA Division Director — Date: 5.18.94

(Block 9 Continued) BASIS OF EVALUATION/DESCRIPTION OF OBSERVATIONS:

During the dates of March 8, 1994, March 15, 1994, and April 21, 1994, a surveillance was performed to determine that the development of a calibration equation was performed by USGS. The following items were verified during the conduct of the surveillance:

1. Verified that the person performing this procedure had been trained and was proficient in USGS Technical Procedure NWM-USGS-HP-254, Revision 0, "Development and Use of a Calibration Equation for a Hand Held Neutron Moisture Meter."

2. Verified that a field calibration equation was developed through a regression analysis performed in accordance with Paragraph 4.2.1 of the above procedure to relate the core water content (as measured through USGS Technical Procedure NWM-USGS-229, "Determination of Water Content and Physical Properties for Laboratory Rock Samples") and the neutron moisture meter counts (as determined through USGS Technical Procedure NWM-USGS-HP-62, "Method for Measuring Sub-Surface Moisture Content Using a Neutron Moisture Meter") for the following boreholes and neutron moisture meters (at time of completion of drilling and core removal) as indicated:

   USW UZ N-27--Meter 3 (2 runs), Meter 6, Meter 7, Meter 8
   USW UZ N-55--Meter 3 (4 runs)
   USW UZ N-54--Meter 3, Meter 6, Meter 7, Meter 8

3. Verified that the above field calibration equation was then used to determine the effective water content of the transfer standards (Tanks 1 and 3 of the Calibration Tanks referred to in procedure HP-62), in accordance with Paragraph 4.2.2 of procedure HP-254. The counts measured in the tanks were converted to effective water content using the regression equation developed for the individual meters.

Personnel contacted:

Alan Flint, USGS
Dave Hudson, USGS
Kevin Ellett, USGS
Kenneth McFall, Quality Assurance Technical Support Services(QATSS)
Thomas Higgins, QATSS

(Block 10 Continued) SURVEILLANCE CONCLUSIONS:

The results of the surveillance are considered to be satisfactory and in accordance with procedure HP-254, Revision 0 and meet the intent of Section 12, Paragraph 12.2.1 A of the Quality Assurance Requirements and Description, DOE/RW-0333P of the Office of Civilian Radioactive Waste Management. There were no deficiencies or discrepancies identified and no Corrective Action Requests were issued as a result of this surveillance.

cc w/encl:
D. A. Dreyfus, HQ (RW-1) FORS
R. W. Clark, HQ (RW-3.1) FORS
W. L. Belke, NRC, Washington, DC
J. W. Gilray, NRC, Las Vegas, NV
R. R. Loux, NWPO, Carson City, NV
Cyril Schank, Churchill County Commission, Fallon, NV
D. A. Bechtel, Clark County Comprehensive, Las Vegas, NV
J. D. Hoffman, Esmeralda County, Goldfield, NV
Eureka County Board of Commissioners,
Yucca Mountain Information Office, Eureka, NV
Lander County Board of Commissioners, Battle Mountain, NV
Jason Pitts, Lincoln County, Pioche, NV
V. E. Poe, Mineral County, Hawthorne, NV
P. A. Niedzielski-Eichner, Nye County, Chantilly, VA
L. W. Bradshaw, Nye County, Tonopah, NV
William Offutt, Nye County, Tonopah, NV
Florindo Mariani, White Pine County, Ely, NV
B. R. Mettam, County of Inyo, Independence, CA
Mifflin and Associates, Las Vegas, NV
S. L. Bolivar, LANL, Los Alamos, NM
R. E. Monks, LLNL, Livermore, CA
W. J. Glasser, REECo, Las Vegas, NV
D. J. Tunney, RSN, Las Vegas, NV
R. R. Richards, SNL, Albuquerque, NM, M/S 1333
R. P. Ruth, M&O/Duke, Las Vegas, NV
T. H. Chaney, USGS, Denver, CO
J. B. Harper, SAIC, Las Vegas, NV
C. J. Henkel, NEI, Las Vegas, NV
C. K. Van House, YMQAD/QATSS, Las Vegas, NV
R. L. Maudlin, YMQAD/QATSS, Las Vegas, NV

TVA

Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37379-2000

Ken Powers
Vice President, Sequoyah Nuclear Plant

May 19, 1994

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

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) UNITS 1 AND 2 - EMERGENCY PREPAREDNESS IMPLEMENTING PROCEDURE (EPIP) REVISION

In accordance with the requirements of 10 CFR 50, Appendix E, Section V, enclosed is a copy of a revision to SQN EPIPs. The revised procedure provided is EPIP-8, Revision 3, "Personnel Accountability and Evacuation." The effective date of the procedural change was April 20, 1994.

If you have any questions concerning these revisions, please telephone C. H. Whittemore at (615) 843-7210.

Sincerely,

Ken Powers

Ken Powers

Enclosure
cc: See page 2

250036

A045
1/1

cc: Mr. D. E. LaBarge, Project Manager (Enclosure)
U.S. Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852-2739

NRC Resident Inspector (Enclosure will be provided by
SQN Document Control Unit)
Sequoyah Nuclear Plant
2600 Igou Ferry Road
Soddy-Daisy, Tennessee 37379-3624

Regional Administrator (Enclosure)
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323-2711

FILING INSTRUCTIONS

EPIP-8

Remove and Destroy			Insert		
Page	Date	Revision	Page	Date	Revision
1-9	06/23/89	2	1-10	APR 20 1994	3
Attachment 1		0	Attachment 1		1

0098h/vbo

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT

EMERGENCY PLAN IMPLEMENTING PROCEDURE

EPIP-8

PERSONNEL ACCOUNTABILITY AND EVACUATION

Revision 3

Quality Related

PREPARED/PROOFREAD BY: Nick S. Catron DATE: 4/20/94

SIGNATURE: Nick S. Cat

RESPONSIBLE ORGANIZATION: Emergency Preparedness

APPROVED BY: [signature] DATE: 4/22/94

EFFECTIVE DATE: 4/20/94

REVISION DESCRIPTION: Deleted reference to Medical Services inside site area. Corrected section titles. Defined OCA as EAB. Changed MPC to DAC.

TABLE OF CONTENTS

1.0 PURPOSE

To describe the method for accounting for all personnel and visitors within the site area prior to an orderly evacuation of a building(s) or the site area during a radiological emergency. This procedure also describes the method for notifying all site personnel and gives guidance for reentry initiation. The term 'onsite' (also called the site area), is the area bounded by the outermost fence surrounding the plant through which access is controlled.

2.0 REFERENCES

2.1 Interface Documents

SNP Personnel Radiological Emergency Accountability System (PREAS) Functional Requirements Document

3.0 INSTRUCTIONS

3.1 Section Supervisors

A. Section supervisors or designated representatives are responsible for ensuring each individual in their section/organization is trained and knows where to report when a radiological emergency or accountability occurs.

B. Section supervisors or their assigned alternate shall ensure that all personnel in their section have swiped the card reader in their particular assembly area when the accountability sirens are sounded.

C. Supervisors will maintain contact with Nuclear Security in order that evacuation of non-essential personnel can be accomplished if required. Non-essential personnel are those who have not received Radiological Emergency Plan training, unless otherwise designated by the Site Emergency Director (SED) or are not on the emergency access list in the main access portal.

3.2 Individuals

A. Upon hearing the emergency siren, all persons within the site area will go to their preassigned assembly areas and swipe their site area badge through the card reader and wait for further instructions. Individuals temporarily assigned to areas or sections will be accounted for in that section or area. In the event that a card reader will not read a site area badge contact NS/PS (6144 or 6226) to be accounted for.

B. If individuals are wearing protective clothing and working in a contaminated zone, they should remove the outer protective articles (shoe covers, gloves, outer coveralls) at the stepoff pad and proceed to the 690' RADCON lab assembly area and swipe the card reader. To prevent the possible spread of contamination, these individuals should:

1. Walk at the outer edge of a normal passage route.

2. Avoid contact with other individuals.

3. Request RADCON surveillance as soon as possible.

C. All plant employees that have preassigned work stations during an emergency and who are escorting visitors (power block visitor badge), upon hearing the emergency sirens, should escort their visitors to an assembly area, swipe the card reader if that visitor is to remain in the protected area and assign them to an escort prior to proceeding to their own work station. Those visitors will remain in that assembly area and await instructions. Visitors (site area visitor badge) not inside the protected area will be sent to the site area access control portal where they will be accounted for.

D. After visitors inside the site area have been accounted for, at the direction of the SED they may be allowed to leave; however, RADCON personnel will check all people and vehicles for contamination prior to their release offsite.

3.3 Site Security Manager

A. The Site Security Manager is responsible for generating assembly area accountability reports from the PREAS and for reporting accountability results to the SED.

B. The responsible person in the Secondary Alarm Station (SAS) will provide the Site Security Manager in the TSC the PREAS generated accountability report information for all personnel assembled in all assembly areas.

C. If there are persons who cannot be located within about 30 minutes, the TSC Security Manager, with SED approval, will dispatch a search and rescue team from the Operations Support Center (OSC) to locate those personnel.

D. Accountability is considered to be complete when all personnel have been accounted for or are known by name if not accounted for.

3.4 Emergency Response Employees Having Operational Assignments During Radiological Emergencies

Upon hearing the emergency sirens, the following people that are assigned to the following organizations will respond as follows:

3.4.1 Operations

All Operations Section personnel outside the protected area will assemble at the OSC. The U-1 ASOS will proceed to the OSC swipe the card reader and fill the Operations Advisor position. If the U-1 ASOS is not able to fill the OSC Advisor position, the SOS will designate or call in a person to fill the position.

All Operations personnel inside the protected area not assigned to the control room will secure the operation in which they are engaged and proceed to the main control room card reader, swipe through, and report to the Operations section lunchroom or main control room for further instructions. The Shift Operations Supervisor (SOS) will assign an individual to ensure persons in the main control room have all swiped the main control room card reader.

3.4.2 Radiological Control Technicians

Proceed to the RADCON lab in the service building, elevation 690', and swipe the card reader and stand by for instructions.

3.4.3 Site Security

All Nuclear Security personnel shall secure all doors and gates as required, report for assigned duties, make an accountability of themselves, direct all visitors within the Site Area to the site access control portal, and stand by for further instructions.

Site Security will dispatch officer(s) to remote and high noise locations within the Site Area to notify personnel of emergency conditions and direct them to seek shelter.

Exclusion Area Boundary

Upon hearing the emergency siren, Nuclear Security will secure all normal access to TVA property after 5 minutes and advise TVA employees outside the site area but within the exclusion area boundary to seek shelter.

All visitors (civilians) outside the site area will be escorted to the exclusion area boundary and released. Affected areas of Chickamauga Lake will be evacuated following notification of the State by TVA.

3.4.4 Technical Support Center (TSC) Personnel

Proceed immediately to the main control room card reader, swipe through, and continue to the TSC. If the TSC is staffed, the SED will assign an individual to ensure all persons in the TSC have swiped the main control room card reader.

3.4.5 Operation Support Center (OSC) Personnel

Proceed immediately to the OSC, after swiping a card reader. If the OSC is staffed, the OSC Manager will assign an individual to ensure all persons in the OSC have swiped a card reader.

3.4.6 Radiochemical Laboratory Personnel and Radiochemical Lab Supervisor

Report to the Radiochemical Laboratory and swipe the card reader on elevation 690' near the RADCON lab and stand by for instructions.

3.4.7 Fire Brigade/MERT

Proceed immediately to the 706' elevation fire cages with turnout gear and equipment, swipe the card reader, and standby for instructions. If response to a fire or medical emergency is already underway, the Incident Commander is the responsible individual to account for the Fire Brigade to the SAS (6144 or 6226) or OSC.

3.4.8 Damage Control/Assessment Teams

If dispatched, contact the OSC for instructions. Consider if continued response is necessary to mitigate emergency conditions, or protect the public health and safety, and the response team is not threatened. If continued response is necessary, the OSC is responsible to account for the team(s) members.

3.5 All Other Plant Employees and Visitors Not Involved in Operational Activities During Radiological Emergencies

A. Plant employees proceed to the nearest card reader, preferably their assigned assembly areas, as listed in Attachment 1 or as required by Section Supervisors and swipe the card reader for accountability.

B. Visitors will be required to proceed to the site area access control portal and Nuclear Security will account for them. Exceptions for visitors are noted in paragraph 3.2.3.

C. When accountability is complete, individuals will remain in the assembly areas. Subsequent activity or movement of personnel will be controlled by direction from the SED or Nuclear Security.

3.6 Total Plant Evacuation

If the SED deems it necessary, or if radiation levels at an assembly point would cause a radiation dose of 100 mrem in one hour or if airborne radioactivity is in excess of 10CFR20 derived air concentration (DAC) limits, he, using the public address system and Nuclear Security, will order evacuation to the employee parking lot or instruct Nuclear Security to complete the evacuation if radiation levels at the employee parking lot are unsafe for occupancy. All personnel exiting the site area shall be required to swipe the gatehouse card reader.

In the event of a total plant evacuation of non-essential personnel, the assembly point may be moved to the Power Services Center at Chickamauga Dam after passing a RADCON check point. The SED will ensure all personnel and vehicles pass through a RADCON check point prior to being released. Instructions will be given by the Site Emergency Director, based on local radiation and contamination conditions. He may recall evacuated people as needed. The CECC Director will be notified of the intended evacuation.

If the personnel require transportation and/or sheltering, the SED will coordinate arrangements for needed assistance with the CECC. If the evacuees require radiological decontamination, they will be informed of transportation, sheltering, and decontamination arrangements prior to leaving the site area. The primary evacuation shelter for onsite contaminated personnel will be Watts Bar Nuclear Plant approximately 50 miles north of Sequoyah Nuclear Plant. Evacuation of onsite non-contaminated individuals would take place along one of the three southbound evacuation routes to the Chattanooga Power Services Center at Chickamauga Dam. The preferred route would be determined following discussion with the CECC or Hamilton County EOC and would depend upon traffic conditions, road (weather) conditions, and radiological hazards. RADCON personnel from the plant site, Watts Bar Nuclear Plant or CECC will respond to the Watts Bar Nuclear Plant shelter area to support personnel decontamination activities if there is a need.

Upon orders from the SED, the training center Building Emergency Coordinator or senior employee present shall evacuate the training center personnel in accordance with established procedures.

3.7 Particular Area Evacuation

In the case where only a particular area has been evacuated, the RADCON Superintendent will be notified of all relocations, and personnel will respond as follows:

3.7.1 All Persons Within the Affected Area

Evacuate to a safe area as defined by RADCON personnel. Swipe the card reader in the alternate assembly area upon arrival and remain there for further instructions.

3.7.2 Fire Brigade/MERT

Contact the OSC/SOS to determine the nature of the incident and receive any instructon. If not responding, standby for further instructions.

3.7.3 All Plant Employees and Escorted Visitors Not In Affected Area

All plant employees and escorted visitors not in the affected area will continue assigned tasks, unless instructed otherwise.

3.8 Plant or Area Reentry

As soon as possible after personnel evacuation has been accomplished, instructions will be initiated to restore the plant to normal conditions. However, before any reentry is attempted, complete radiological surveys will be made if the cause of the evacuation is radiological in nature. The SED will authorize reentry only when he is assured that the emergency has been controlled.

3.9 PREAS Failure

In the event the PREAS system is unable to accomplish its designed function Nuclear Security will implement the following steps as compensatory measures:

A. Nuclear Security will account for all badges issued for access into the protected area.

B. Nuclear Security will conduct a search of the Turbine Building to determine if there are injured personnel in the area.

SOURCE NOTES

REQUIREMENTS STATEMENT	SOURCE DOCUMENT	IMPLEMENTING STATEMENT
	NP Radiological Emergency Plan (NP-REP)	

1.0 REVISION PROCESS

Emergency Preparedness Planning Manager

A. **MAINTAIN** a listing of PREAS Card Reader locations by building and reporting groups that will require review by the Emergency Preparedness Staff in this attachment as follows:

1. **COLLECT** information on personnel and accountability cart locations.

2. **EVALUATE** personnel and cart availability by locations.

3. **SUBMIT** a markup of this attachment with the changes to SPS.

SPS

B. **PROCESS** the update per SSP-2.3, Appendix F.

C. **RETURN** to the Emergency Preparedness Planning Manager

Emergency Preparedness Planning Manager

D. **SIGN** concurrence signature below.

E. **RETURN** to SPS.

SPS

F. **FORWARD** to DCRM per SSP-10.2.

DCRM

G. **DISTRIBUTE** per SSP-10.2.

Nick G. Cot / 4/20/94
Emergency Preparedness Planning Manager / Date

LOCATION	REPORTING GROUP(s)
O&PS-1	Power stores employees and other employees located on O&PS-1
O&PS-2	All employees located on O&PS-2.
O&PS-3	All employees located on O&PS-3.
O&PS-4	Site Vice President's Staff and other employees located on O&PS-4.
SB-1	
a. South Hallway- - - - - - - - - - -	Employees located in SB-1.
b. Main Entrance - - - - - - - - - -	SB trailers and other employees located on SB-1.
c. MODs Shop - - - - - - - - - - - -	MOD Craftsman, insulators, painters, shop personnel, and carpenters.
SB-2 - - - - - - - - - - - - - - - - - - -	Outage group, daily scheduling, planners, and other groups located on SB-2.
690' RADCON Lab	RADCON and Chemistry
732' CB (SOS Clerk's Office)	TSC personnel and OPS personnel

LOCATION	REPORTING GROUP(s)
POB-2	Plant Manager's Staff
El. 706' Service Bldg Breezeway	OSC personnel, Radwaste group, cafeteria and all other groups located in the POB or Service Building not specifically assigned.
Instrument Maintenance Shop	Instrument Maint. employees
Electrical Maintenance Shop	Electrical Maint. employees
Mechanical Maintenance Shop	Mechanical Maint. employees, and craftsmen.

NOTE: In the event an individual cannot report to his designated card reader within 15 minutes he/she should swipe the nearest card reader and remain in that area.

Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402-2801

10 CFR 50.54(q)

May 20, 1994

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

Gentlemen:

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296
		50-327
		50-328

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 1, 2, AND 3 - SEQUOYAH NUCLEAR PLANT (SQN) - UNITS 1 AND 2 - REQUEST FOR NRC REVIEW AND APPROVAL OF COST BENEFICIAL LICENSING ACTION - PROPOSED EMERGENCY PLAN CHANGE - CORE/THERMAL HYDRAULIC ENGINEER STAFF AUGMENTATION

TVA requests approval of a change to the Radiological Emergency Plan (REP). The proposed change revises the manner in which TVA would provide expertise in the area of core/thermal hydraulic analysis during the first hour of an emergency. This request is submitted pursuant to the requirements of 10 CFR 50.54(q).

Guidance for supplying core/thermal hydraulic expertise is contained in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Table B-1, "Minimum Staffing Requirements for NRC Licensees for Nuclear Power Plant Emergencies." In the functional area of Plant System Engineering, Repair, and Corrective Actions, NUREG-0654 specifies that an individual having expertise in core/thermal hydraulic analysis be available to augment the on-shift staff within 30 minutes of an emergency.

A045

In order to provide core/thermal hydraulic expertise within the 30-minute time limitation, TVA must assign engineers to on-site around-the-clock shift coverage at BFN and provide for engineers to be on call at SQN. Rather than providing this level of coverage, TVA considers that the on-shift Shift Technical Advisors (STAs) can perform needed core/thermal hydraulic assessments during the first 60-minutes of an emergency. Therefore, TVA proposes to revise the REP to specify that an individual having expertise in core/thermal hydraulic analyses be available to augment the on-shift staff within 60 minutes as the Technical Support Center is staffed and activated.

The STAs are qualified and have the necessary education, background, and training for performing core/thermal hydraulic assessments. Responsibility for performing these assessments would not impact the STA's ability to respond during an emergency since the STAs perform core/thermal hydraulic assessments as a routine part of their on-shift and emergency assessment duties.

TVA has determined that the proposed change represents a cost beneficial licensing action (CBLA) since the 30 minute staff augmentation requirement involves a high cost and provides a low safety benefit. TVA has determined that requiring the on-shift core/thermal hydraulic engineering coverage at BFN often results in significant productivity losses. In order to increase productivity, TVA proposes to revise the REP thus eliminating mandatory shift coverage at BFN. TVA estimates that the proposed REP change, if approved, will result in cost benefits of approximately $1.68 million over the remaining life of the facility.

There is little safety benefit in requiring that an engineer be available within 30 minutes to provide core/thermal hydraulic expertise. Since the NUREG guidance was written, several improvements in the personnel qualification requirements and instrumentation provided for accident assessment have been implemented. The training and qualification requirements for STAs are much more stringent than in the past. Additionally, instrumentation is now available (e.g., Safety Parameter Display System) that facilitates analysis of core/thermal hydraulic performance. These improvements have made it no longer necessary to provide core/thermal hydraulic engineering augmentation within 30 minutes.

In order to maintain consistency throughout TVA's nuclear units, the 30 minute staffing requirement will also be revised for SQN. Enclosure 1 provides a detailed discussion to support TVA's request. Enclosure 2 provides a marked-up copy of BFN REP Table A-1, "Site Emergency Organization," and SQN REP Figure B-1, "Technical Support Center Emergency Preparedness," to show the proposed change. There are no commitments made in this letter. If there are any questions regarding this submittal, please telephone me at (615) 751-2687.

Sincerely,

Sydney W. Spencer

for Roger W. Huston
Manager
Nuclear Licensing and Regulatory Affairs

Enclosures
cc: See page 4

Enclosures
cc (Enclosures):

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NRC Resident Inspector
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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 AND 2

OPERATING AND MAINTENANCE COST REDUCTION PROGRAM
COST BENEFICIAL LICENSING ACTION
CHANGE TO EMERGENCY PLAN STAFFING REQUIREMENT
CORE/THERMAL HYDRAULIC ENGINEER STAFF AUGMENTATION

I. SUMMARY OF CBLA

TVA proposes to revise the manner in which supplemental core/thermal hydraulic engineering support is provided during the first hour of an emergency. In order to meet current TVA Radiological Emergency Plan (REP) 30-minute time limitations for providing core/thermal hydraulic staff augmentation during emergencies, engineers from the Technical Support organization are assigned to around-the-clock shift coverage at BFN and are required to be on call at SQN. Needed core/thermal hydraulic expertise can be provided by the on-shift Shift Technical Advisor (STA); therefore, TVA proposes to revise the REP to specify that supplemental core/thermal hydraulic expertise be provided within 60 minutes as the Technical Support Center (TSC) is staffed and activated.

The change for BFN is requested since TVA would like to remove the Technical Support engineers from mandatory shift coverage. This will result in significant productivity enhancements and associated cost savings. The change for SQN is requested to maintain consistency at TVA nuclear sites.

TVA considers that the proposed change to the REP meets the standards of 10 CFR 50.47(b) and the requirements of 10 CFR 50 Appendix E. However, TVA recognizes that this change could be considered a decrease in the effectiveness of the REP since staff augmentation would be accomplished in 60 minutes following an emergency declaration rather than the current 30 minutes. Accordingly, TVA is requesting NRC approval of this proposed change pursuant to the provisions of 10 CFR 50.54(q).

II. BACKGROUND

The REP prescribes the overall measures used for responding to emergency situations at TVA's nuclear units. The REP is modeled after the guidance contained in NUREG 0654, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." In the functional area of providing technical support during emergencies, NUREG 0654 Table B-1, "Minimum Staffing Requirements for Emergencies," specifies that an engineer having expertise in the area of core/thermal hydraulics be available to augment on-shift personnel within 30 minutes of the emergency. In order to provide core/thermal hydraulic expertise within the 30-minute time limitation, TVA elected to meet the core/thermal hydraulic support augmentation commitment using engineers from the Technical Support organization. At BFN, TVA must use engineers who are on-site around-the-clock. At SQN, TVA must use engineers who are on a limited response time call out.

If an emergency occurs at BFN, the assigned technical support engineer is responsible for reporting to the on-shift STA and providing assistance as needed. The technical support engineer continues to report to the STA until the technical support center (TSC) is activated. When the TSC is activated, the technical support (core/thermal hydraulics) engineer reports functionally to the Technical Assessment Team Leader in the TSC.

If an emergency occurs at SQN, the on-shift STA provides core/thermal hydraulic support until the technical support engineer arrives on-site. At this time the technical support engineer assumes these duties and reports to the STA until the TSC is activated. When the TSC is activated, the technical support engineer will the report to the TSC Technical Assessment Team Leader.

The engineers chosen to provide core/thermal hydraulic support are qualified prior to assuming their assigned duties. Core/thermal hydraulics engineer duties are encompassed by the Technical Assessor duties specified in BFN's and SQN's Emergency Plan Implementing Procedure(EPIP)-6, "Activation and Operation of the Technical Support Center (TSC)." These duties include:

- Complete trend graphs as needed.
- Provide the TSC staff and Central Emergency Command Center (CECC) Plant Assessment Team with current assessments on plant conditions.
- Project future plant status based on current conditions.
- Provide technical support as needed.
- Provide information for the TSC status boards.

At BFN, technical support engineers were originally selected since they had been assigned around-the-clock shift coverage to support maintenance and operations activities. Since Unit 2 has been restarted and is operating, the need to provide around-the-clock shift coverage to support maintenance and operations is no longer necessary. Therefore, TVA would like to eliminate mandatory shift coverage to improve productivity.

III. REQUESTED NRC LICENSING ACTION

Pursuant to the requirements of 10 CFR 50.54(q), TVA requests approval of a change to the current REP requirement that supplemental core/thermal hydraulic engineering support be available to augment the on-shift staff within 30 minutes. TVA proposes to change the staff augmentation requirement from 30 to 60 minutes and denote that the STA is responsible for performing core/thermal hydraulic assessments during the first 60 minutes of an emergency. Enclosure 2 provides marked-up copies of BFN Figure A-1, "Site Emergency Organization," and SQN Figure B-1, "Technical Support Center Emergency Preparedness," to show the proposed change.

IV. BASIS FOR REQUESTED NRC LICENSING ACTION

The intent of the NUREG 0654 guidance to have a core/thermal hydraulic engineer available within 30 minutes is to ensure that sufficient expertise in this area is readily available. TVA considers that the STAs provide adequate expertise for assessing core/thermal hydraulic performance during the first 60 minutes of an emergency for the following reasons:

- STAs are on-shift and in position to assess core/thermal hydraulic performance during the first stages of an accident.
- STAs have the education and experience necessary for performing core/thermal hydraulic assessments.
- STAs receive extensive training to ensure that they are qualified to perform core/thermal hydraulic assessments.
- STA duties and responsibilities require that they perform core/thermal hydraulic assessments.
- Instrumentation is now available, which was not installed when the NUREG guidance was written, to facilitate analysis of core/thermal hydraulic performance (e.g., Safety Parameter Display System [SPDS]).

STAs are part of the on-shift staff since they serve as an on-shift technical advisor to the Shift Operations Supervisors. During the first stages of an emergency requiring core/thermal hydraulic assessments, the STAs are aware of the plant status and precursor situations that may have led to the emergency. The STAs are in position to perform needed core/thermal hydraulic assessments.

The background, education, and experience requirements for STAs are adequate for providing core/thermal hydraulic expertise. STAs must have a bachelor's degree or equivalent in engineering or an applied science. STAs must have at least one year of nuclear power plant experience and six months of this experience must be on site. STAs must be knowledgeable of control room instruments and controls.

STAs receive extensive training that qualifies them to perform core/thermal hydraulic assessments. The training includes basic theoretical and advanced instruction. STA training related to assessing core/thermal hydraulic performance is provided in the following areas:

- Basic engineering principles, including plant specific thermodynamics/fluid flow, reactor physics, system engineering, plant response, and instrumentation.

- Plant transient and accident response.

- Mitigating core damage.

The primary objective of the STA during emergencies is to maintain an overview of the event by observing plant parameters and their trends, operator responses and actions, system responses and performance, and plant environmental conditions through comparison of actual and predicted conditions. In order for the STA to meet this objective, the STA routinely assesses core/thermal hydraulic performance. Table 1 lists the duties and responsibilities of the STA during emergency conditions. The duties and responsibilities that require performance of core/thermal hydraulic analyses are noted.

The guidance provided in NUREG 0654 was published in 1980. Since then, improvements in accident monitoring capability have been implemented that greatly enhance the ability to monitor core/thermal hydraulic performance. Of particular use for assessing plant performance in this area is the SPDS. The SPDS is designed to automate and enhance the generation of information necessary for rapid detection and evaluation of abnormal and emergency conditions.

TVA has reviewed the STA duties and responsibilities and determined that the STA is fully capable of performing necessary core/thermal hydraulic assessments during the first 60 minutes of an emergency. The support provided by the Technical Support engineer is not needed to ensure that the STA's duties and responsibilities are executed.

Support for the core/thermal hydraulic assessment function will be readily available within 60 minutes as the TSC is staffed and activated. The proposed revision will specify that individuals with expertise in performing core/thermal hydraulic assessments be part of the emergency response organization and report to the TSC within 60 minutes of an emergency being declared.

Finally, the duties and responsibilities of the core/thermal hydraulic engineers related to their role in the TSC (e.g., providing the TSC and CECC staffs with current plant conditions) are not needed during the first hour of an emergency. The TSC and CECC are not required to be operational during the first hour of an emergency. Therefore, delaying the response time of the core/thermal hydraulic engineer to 60 minutes will not impact the ability of TVA to meet TSC performance objectives.

V. JUSTIFICATION FOR HIGHER PRIORITY REVIEW

A. CBLA Is Safety Neutral

TVA considers that the proposed change is consistent with the provisions of 10 CFR 50.47(b), 10 CFR 50 Appendix E, and the intent of NUREG 0654. The proposed change does not affect the ability to ensure that necessary core/thermal hydraulic expertise is available. As described above, the STA is capable of providing adequate core/thermal hydraulic engineering support during the first hour of an emergency. STAs routinely perform core/thermal hydraulic assessments as part of their emergency duties. The STA is more qualified, better trained, and in a better position to make these assessments than an individual reporting within 30 minutes.

Extending the time required for the core/thermal hydraulic engineer from 30 minutes to 60 minutes would not affect the operation of the TSC. The TSC is not required to be staffed and operational during the first hour of an emergency. An engineer capable of performing core/thermal hydraulic assessments would be readily available when the TSC is staffed and operational.

B. CBLA Provides Significant Cost Savings and Other Benefits

TVA conservatively estimates that approval of this revision to the REP would result in indirect cost savings of approximately $1.68 million over the remaining life of the facility. The cost savings would be realized since inefficiencies could be eliminated that are associated with providing around-the-clock shift coverage at BFN. While on shift work, TVA estimates that a significant amount of an engineers time is lost due to the following concerns:

- Routine business and engineering decisions are typically made by or approved by supervisory/management personnel. On weekends and back-shifts, these personnel are generally unavailable or it takes extra effort to contact the appropriate individual.

- Engineers frequently require support of personnel that are not assigned shift coverage. Productivity is often lost since "support" personnel are not readily available.

- Shift coverage results in some system engineers not being available during normal business hours to perform emergent work related to their normal responsibilities. This often results in situations where the engineer's backup must respond to an issue. Time is wasted because the backup engineer must expend more time to become familiar with the system and also, the backup engineer's normal responsibilities get delayed.

- Some productivity is lost due to the "human factors" associated with working on backshifts and providing rotating shift coverage. The engineers assigned shift coverage normally work day-shift.

- Providing shift coverage requires that a certain amount of overtime be dedicated to this function. Due to overtime restrictions, it is often difficult to provide engineering coverage for emergent situations.

- Time spent on shift-turnover would be eliminated.

In addition to the above, significant management attention is needed to administer the scheduling and contingencies (e.g., calling out another individual if the assigned engineer is unable to report to work) associated with providing shift coverage.

TVA conservatively estimates that productivity improvements of approximately 15% will occur by eliminating shift coverage. Additionally, elimination of the requirement to maintain shift coverage for emergency response purposes would allow improved flexibility in providing the appropriate level of staffing to support plant operations and maintenance. Staffing levels could be promptly adjusted to provide the maximum benefit for changing circumstances. Thus, TVA could eliminate shift coverage to improve productivity and allow increased management attention to other, more important work.

Providing around-the-clock shift coverage involves a significant expenditure of resources. A minimum of three engineers per day (one per 8-hour shift), seven days per week are needed. Over 22 years (the Unit 2 operating license expires in 2016), a 15% productivity increase results in savings of approximately $1.68 million at today's salary costs.

VI. CONCLUSION

The proposed change is consistent with the provisions of 10 CFR 50.47(b) and 10 CFR 50 Appendix E. The STA is fully qualified and capable of assessing core/thermal hydraulic performance during the first 60 minutes of an emergency. The STA's performance in this capacity meets the intent of NUREG 0654 guidance to have engineering support for assessing core/thermal hydraulic readily available.

The responsibilities of the core/thermal hydraulic engineer that relate to his/her function in the TSC (e.g., providing the TSC and CECC staffs with current plant conditions) are not needed for the first 60 minutes. As the TSC is staffed and becomes operational, an engineer having expertise in core/thermal hydraulic assessments will be available to perform any necessary support functions.

TVA has determined that the proposed PEP change is a cost beneficial licensing action since it is safety neutral and provides significant cost savings. The proposed change will allow TVA to make changes that are intended to improve productivity as well as provide flexibility in meeting changing plant conditions. Therefore, TVA requests expeditious NRC review of this proposed exemption.

TABLE 1

STA DUTIES AND RESPONSIBILITIES DURING EMERGENCY CONDITIONS

STA EMERGENCY DUTIES AND RESPONSIBILITIES	THERMAL HYDRAULIC ASSESSMENT REQUIRED?
Maintain an overview of the event by observing plant parameters and their trends, operator responses and actions, system responses and performance, and plant environmental conditions through comparison of actual and predicted conditions.	Yes
Verify that systems and team actions are responding and serving to correctly mitigate the event, and advise of departures from the expected and required.	Yes
Effectively communicate event mitigation information to the appropriate organizations as directed.	No
Assist in interpreting and applying the Technical Specifications, including recognition and notification of plant conditions which involve Safety Limits or Limiting Safety System Settings.	No
Provide independent verification of critical safety functions.	Yes
Recognize events that may affect the safety, health, and/or welfare of the public or plant personnel.	Yes
Recognize events that may affect plant security.	No
Recognize events that require prompt NRC notification and/or result in Emergency Plan Implementing Procedure entry conditions.	No
Recognize plant conditions that are outside of analyzed conditions and may lead to core damage.	Yes
Perform calculations, as necessary, for assessment, evaluation, and compliance.	Yes
Monitor plant critical parameter indications for operability and if parameters become unavailable due to instrument failure, calculate or otherwise determine approximate values for the parameters in question.	Yes
Maintain a chronological log of major events, observations, and recommendations.	No
Assist in determining emergency classification and reporting requirements.	No

STA DUTIES AND RESPONSIBILITIES DURING EMERGENCY CONDITIONS

STA EMERGENCY DUTIES AND RESPONSIBILITIES	THERMAL HYDRAULIC ASSESSMENT REQUIRED?
Recognize plant conditions that effect implementation of the Emergency Operating Instructions. Verify appropriate procedural implementation.	Yes
Determine the appropriate reactor water level instrument(s) to use for a given set of plant conditions.	Yes
Verify the proper operation of the core spray pumps and residual heat removal pumps when they are used.	Yes
Determine if suppression pool, drywell, and reactor pressure vessel temperatures, pressures, and levels can be maintained within safe limits. Provide recommendations based on values and trends.	Yes
Determine the heat capacity level limit and verify suppression pool level can be maintained within safe limits. Provide recommendations based on values and trends.	Yes
Recognize indications and alarms for area temperatures that exceed normal and safe limits. Provide recommendations based on values and trends.	No
Recognize indications and alarms for area radiation levels that exceed normal and safe limits. Provide recommendations based on values and trends.	No
Recognize indications and alarms for area water levels that exceed normal and safe limits. Provide recommendations based on values and trends.	No
Verify, monitor, and log containment water level.	No
Recognize when plant conditions require entrance into the Safe Shutdown Instructions.	No

ENCLOSURE 2

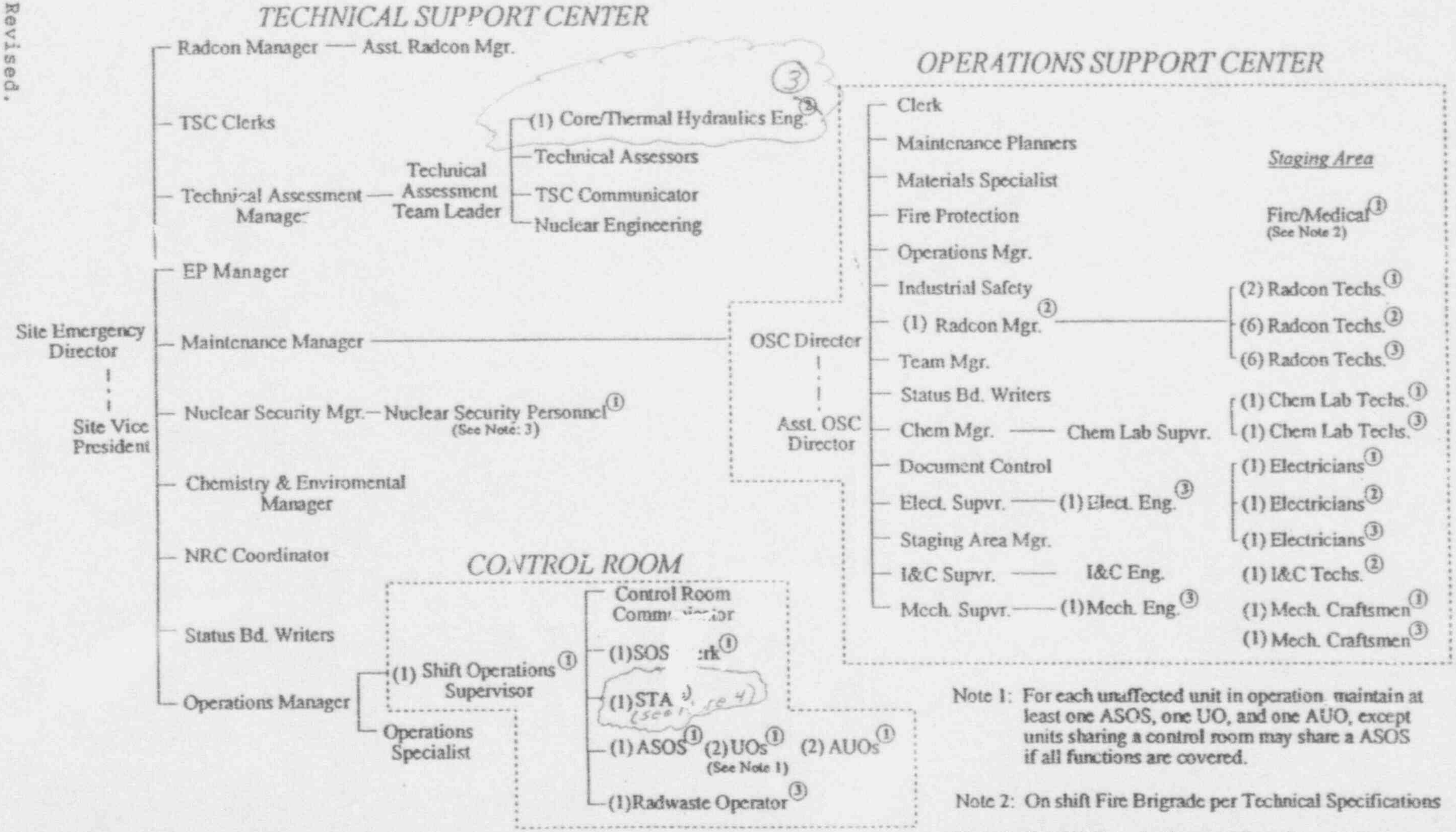
TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3
SEQUOYAH NUCLEAR PLANT
UNITS 1 AND 2

MARKED-UP BFN AND SQN REP
FIGURE A-1
SITE EMERGENCY ORGANIZATION

(see attached)

Entire Page Revised.

FIGURE A-1
SITE EMERGENCY ORGANIZATION
(Including Minimum Staffing and Staff Augmentation)



① On Shift
② Augment within approximately 30 minutes following the declaration of Alert or higher classification
③ Augment within approximately 60 minutes following the declaration of an Alert or higher classification.

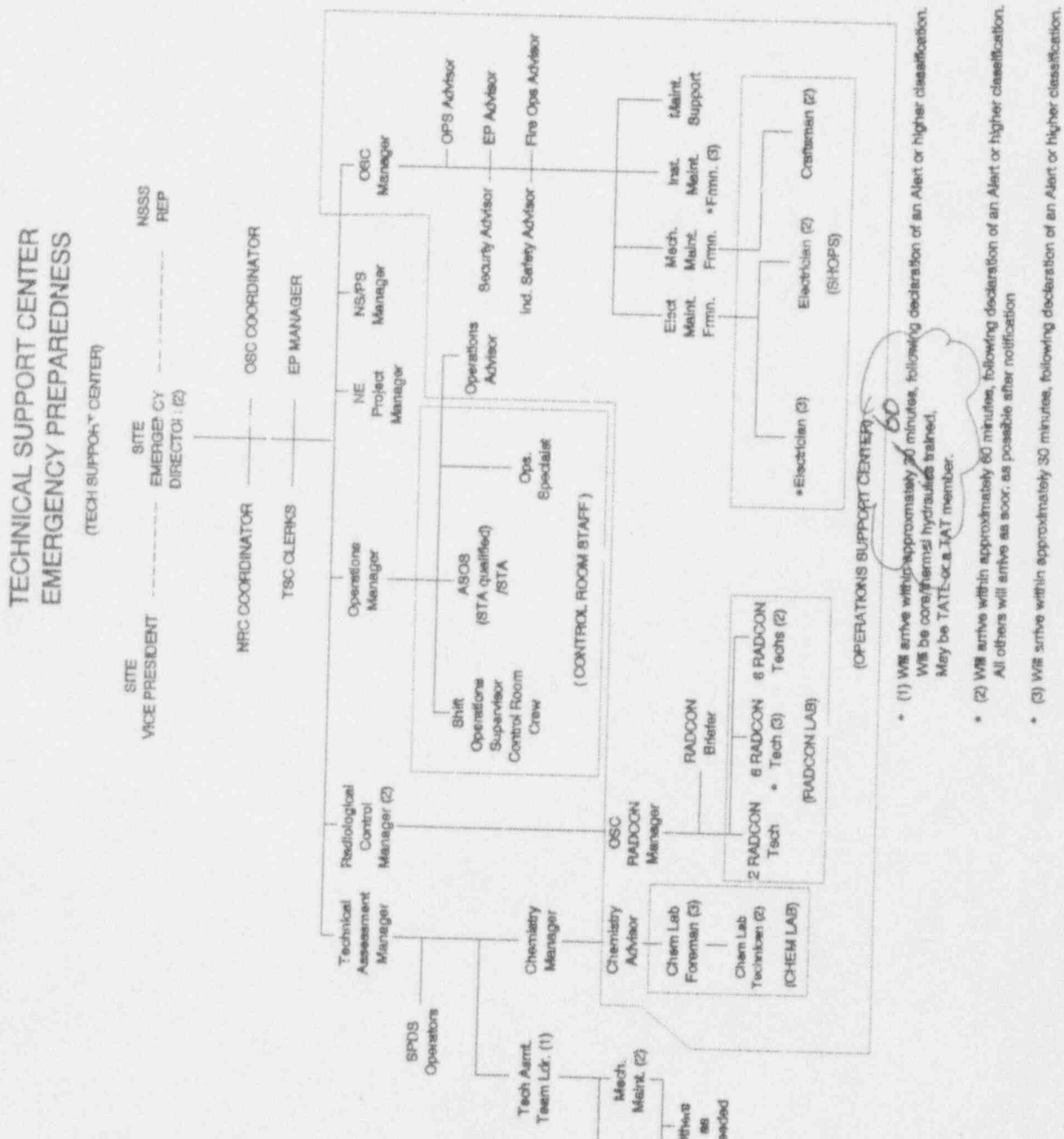
Note 1: For each unaffected unit in operation, maintain at least one ASOS, one UO, and one AUO, except units sharing a control room may share a ASOS if all functions are covered.

Note 2: On shift Fire Brigrade per Technical Specifications

Note 3: On shift Security Personnel per Security Plan

Note 4: Assumes Responsibility for Core/Thermal Hydraulic Assessment function During First 60 minutes of Event

FIGURE B-1



*Revision

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

MAY 23 1994

SCDB:TOM
07100384

The DuPont Merck Pharmaceutical Company
ATTN: Mr. Raymond A. Shepard
Director, Billerica Site
331 Treble Cove Road
North Billerica, MA 01862

Dear Mr. Shepard:

SUBJECT: NRC INSPECTION REPORT NO. 94206
NOTICE OF VIOLATION, NOTICE OF NONCONFORMANCE

This refers to the U.S. Nuclear Regulatory Commission inspection performed by Messrs. T. Matula and S. O'Connor of this office on March 30, 1994.

The inspection consisted of an examination of the implementation of your Quality Assurance Program Approval No. 0384. The inspection included examination of procedures and representative records, and interviews with personnel at your North Billerica, Massachusetts, facility. The preliminary findings were discussed with you and your staff at the conclusion of the inspection.

It was found that the implementation of your Quality Assurance Program failed to meet NRC requirements. The specific findings and references to pertinent requirements are identified in the enclosed Inspection Report, Notice of Violation, and Notice of Nonconformance.

I note that in a letter dated April 7, 1994, Mr. D. Dumas of your staff submitted additional information regarding the three issues listed below which were identified during the inspection. During a telephone conversation with Messrs. S. Roy and R. Greaves of your staff, Mr. T. Matula, NRC Inspection Team Leader, provided the following information:

1. "Applicability of 10 CFR Part 21" - The issue identified during the inspection is an internal procedure that controls deviations and failures to comply, as required by 10 CFR § 21.21, was not available to the inspection team. The information provided in the subject letter pertains to 10 CFR § 21.31, "Procurement documents;" therefore, it does not address the inspection finding.

2. "Calibration of torque wrench" - The issue identified during the inspection is the use of an uncalibrated torque wrench to secure the lid bolts for the Model No. CNS 4-85 packaging for transportation. In the subject letter, it is stated that the torque wrench was sent out for calibration on April 4, 1994; however, the torque wrench was not calibrated at the time of inspection and, therefore, the inspection finding stands. Your timely corrective action regarding this issue is noted.

NT03 1/1

3. "User Check-Off Sheet signed twice by same individual" - The inspection team noted, during the inspection, that the same individual signed an inspection sheet in two locations; once as the inspector and again as the inspector's supervisor. In the subject letter, it is stated that the owner's procedure for preparing this container does not specify that these functions be performed by separate individuals. This has been noted and, therefore, this issue has been resolved.

Please provide us, within 30 days from the date of this letter, a written statement in accordance with the instructions specified in the enclosed Notice of Violation and Notice of Nonconformance (Notices). We will consider extending the response time if you can show good cause for us to do so.

The responses directed by this letter and the enclosed Notices are not subject to the clearance procedures of the Office of Management and Budget, as required by the Paperwork Reduction Act of 1980, in Pub. L. No. 96-511. In accordance with 10 CFR § 2.790 of NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

Should you have any questions about this inspection, we will be pleased to discuss them with you.

Sincerely,

Robert Baer

Robert L. Baer, Chief
Source Containment and Devices Branch
Division of Industrial and Medical
Nuclear Safety, NMSS

Enclosures:
1. Inspection Report No. 94206
2. Notice of Violation
3. Notice of Nonconformance

Reviewed by:
E. Kraus, NMSS
05/04/94

G. Bennington, STSB
05/04/94

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

RADIOACTIVE MATERIAL CONTAINER SUPPLIER INSPECTION

ORGANIZATION: The DuPont Merck Pharmaceutical Company
331 Treble Cove Road
North Billerica, MA 01862

CONTACT: Mr. Francis E. Roy, Jr.
Development Health Physicist
Safety and Environmental Affairs
(508) 671-8242

ACTIVITY: Maintenance and use of radioactive material packaging.

QUALITY ASSURANCE PROGRAM APPROVAL NO.: 0384

INSPECTION REPORT NO.: 94206

Inspection Location: North Billerica, MA	Inspection Date: March 30, 1994	On-site Inspection Hours: 14

INSPECTION BASIS AND SCOPE:

A. BASIS: 10 CFR Part 71, 10 CFR Part 21, and Certificate of Compliance Nos. 6244, 6601, 9070 and 9208.

B. SCOPE: To determine whether procedures have been established, documented, and executed that meet the requirements of 10 CFR Part 21 and 10 CFR Part 71. To determine if the procedures fulfill commitments made in the NRC-approved Quality Assurance Program.

<table>
<tr><td colspan="3">RADIOACTIVE MATERIAL CONTAINER SUPPLIER INSPECTION
INSPECTION REPORT NO.: 94206</td></tr>
<tr><td colspan="3">FINDINGS:
A violation with the requirements of 10 CFR § 21.21 was identified.
Nonconformances with the requirements of 10 CFR Part 71, Sections 125 and 137, were identified.</td></tr>
<tr><td>INSPECTION TEAM LEADER:</td><td>Thomas O. Matula

Thomas O. Matula
NMSS</td><td>DATE: 5/12/94</td></tr>
<tr><td>INSPECTOR:</td><td>Stephen O'Connor

Stephen C. O'Connor
NMSS</td><td>DATE: 05-23-94</td></tr>
<tr><td>REPORT APPROVED BY:</td><td>Robert Baer

Robert L. Baer, Chief
Source Containment and Devices Branch
Division of Industrial and Medical Nuclear Safety, NMSS</td><td>DATE: 5/23/94</td></tr>
</table>

1. SUMMARY

A U.S. Nuclear Regulatory Commission inspection team conducted an announced inspection of The DuPont Merck Pharmaceutical Company (DuPont Merck) in North Billerica, Massachusetts, on March 30, 1994.

The inspection team reviewed DuPont Merck quality assurance (QA) activities to determine whether QA procedures had been established, documented, and executed in fulfillment of commitments made in the NRC-approved QA Program. Inspection findings are based on data collected through review of procedures and controls, inspection of selected documents and records, and interviews with personnel.

The inspection team identified a violation of 10 CFR § 21.21 and items of nonconformance regarding 10 CFR Part 71, Subpart H. The items of nonconformance are summarized in Table 1.

Table 1
Items of Nonconformance

10 CFR Part 71, Section	Subject	No. of Findings
125	Control of measuring and test equipment	1
137	Audits	5

2. DETAILS

NRC conducted an announced inspection at the DuPont Merck facility in North Billerica, Massachusetts, on March 30, 1994. The inspection team reviewed documentation representing implementation of the QA functions. The inspection team focused on maintenance activities and use of transportation packagings with NRC Certificates of Compliance (COCs).

The objectives of the NRC inspection were to verify that the QA functions were accomplished in compliance with regulatory requirements and to determine if the DuPont Merck QA program complied with commitments made to NRC.

The inspection team evaluated the performance of QA activities from the perspective of two functional control elements: management and maintenance. The inspection team focused on the QA activities regarding the packaging models listed in Table 2.

Table 2
Packaging Models

COC No.	Model No.
6244	CNS 4-85
6601	CNS 8-120A
9070	N-55
9208	10-142

2.1 Persons Contacted

An entrance meeting was held on March 30, 1994, to present the objectives of the NRC inspection. An exit meeting was held that same day to present the preliminary findings of the inspection. In addition to the NRC inspection team, the individuals present at the entrance and exit meetings are listed in Table 3.

Table 3
Entrance/Exit Meeting Attendees

*	D. Dumas	Manager, Safety and Environmental Engineering
*	R. Greaves	Supervisor, Radiation Protection
*	F. Roy	Development Health Physicist Safety and Environmental Affairs
**	E. DeMaria	Senior Transportation Safety Specialist
**	J. Haepers	Supervisor, Production
**	R. Shepard	Director, Billerica Site

* Present at entrance and exit meetings.
** Present at exit meeting only.

2.2 Management Controls

The inspection of management controls focused on the extent to which DuPont Merck had implemented the QA Program, and the independence and proficiency of QA personnel responsible for fulfilling QA commitments. The inspection team determined QA program effectiveness by reviewing administrative procedures, inspecting QA documents and records, and interviewing staff members.

A violation was identified regarding 10 CFR § 21.21, "Notification of failure to comply or existence of a defect and its evaluation." This section requires the licensee to adopt appropriate procedures to evaluate deviations and failures to comply associated with substantial safety hazards as soon as practicable.

A procedure that fulfills the requirements of 10 CFR § 21.21 was not available to the inspection team.

Nonconformances were identified regarding 10 CFR § 71.137, "Audits." This section requires the licensee to carry out a comprehensive system of planned and periodic audits to verify compliance with all aspects of the QA program and to determine the effectiveness of the program. The audits must be performed by trained personnel; the audit results must be documented and reviewed by management having responsibility in the area audited; and follow-up action, including re-audit of deficient areas, must be taken. The inspection team identified instances where audits were inadequate.

Section 14.0, "Audits," of the DuPont Merck QA Program states the following:

1. "Internal audits must be conducted routinely once every 12 months." Documented evidence that internal audits were conducted in 1991 and 1992 was not available to the inspection team.

2. "Audit checklists must be established prior to conducting an audit." Audit checklists used in the internal audits conducted on October 30 and November 2, 1990, and December 7, 1993, were not available to the inspection team.

3. "Audits must be performed by the Manager, Safety and Environmental Affairs (or designee) and an additional site operations representative with no direct line responsibility for the function audited." The 1990 and 1993 internal audits were performed by one person. The Manager, Safety and Environmental Affairs (or designee) did not participate in the internal audits.

4. "Audits must include an assessment as to how well the QA Program meets regulatory or other requirements." Documentation regarding an assessment of the QA Program in complying with regulatory requirements during the 1990 and 1993 internal audits was not available to the inspection team.

5. "The originator of an audit report or a designated alternate is required to follow an open item until action is taken to satisfy an audit action item. Records of actions taken to achieve resolution are maintained. Follow-up actions are taken to verify corrective action is effective." Documentation regarding follow-up actions taken, corrective action implementation, and determination of corrective action effectiveness resulting from items identified in the 1990 and 1993 internal audit, was not available to the inspection team.

Maintenance Controls

The inspection team reviewed the implementation of the QA program in the performance of packaging maintenance. Maintenance documents were inspected for completeness and adequacy of scheduled maintenance, actual maintenance performed, and QA program controls. Compliance with packaging maintenance, testing, and inspection requirements was verified. Documents were reviewed for completeness, adequacy, and appropriate approvals.

A nonconformance was identified regarding 10 CFR § 71.125, "Control of measuring and test equipment." This section requires the licensee to establish measures to assure that tools used in activities affecting quality are properly controlled, calibrated, and adjusted at specific times to maintain accuracy within necessary limits. The inspection team identified an instance where control of equipment was inadequate.

The primary lid bolts on the Model No. CNS 4-85 packaging require a torque value of 180±18 lb-ft, and the overpack lid bolts require a torque of 90±9 lb-ft. The 0-250 lb-ft torque wrench used to secure the lid bolts for transportation was not calibrated and it was not a controlled piece of equipment.

3. DOCUMENTS REVIEWED

3.1 Procedures

Quality Assurance Program for Radioactive Material Packages, dated December 1990.

SP-01, Incident Investigation Reporting, dated June 10, 1993.

SP-20, Transportation of Radioactive Material, dated October 24, 1988.

SOP No. 07-004-002, Handling of Mo-99 and Xe-133 Radioactive Material Containers, dated December 16, 1990.

SOP No. 07-005-001, Receipt, Monitoring, Storage, and Disposition of Radioactive Materials, dated February 6, 1994.

TR-OP-004, Handling Procedure for Chem-Nuclear Systems, Inc., Transport Cask No. CNS 4-85, COC No. 6244, Revision E, dated March 11, 1991.

3.2 Records, Reports, and Logs

Incident Investigation Log.

Internal Audit Reports, dated February 19, 1991, and December 23, 1993.

Training records for three members of Receiving and Handling Staff.

Transportation Safety Committee Meeting Minutes, dated February 27 and March 19, 1992, and August 25, 1993.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NOTICE OF VIOLATION

The DuPont Merck Pharmaceutical Company
North Billerica, Massachusetts

Docket No. 07100384

During a U.S. Nuclear Regulatory Commission inspection conducted at The DuPont Merck Pharmaceutical Company (DuPont Merck) in North Billerica, Massachusetts, on March 30, 1994, a violation of a certain NRC requirement was identified (refer to Inspection Report No. 94206). In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C, the violation is listed below:

10 CFR § 21.21, "Notification of failure to comply or existence of a defect and its evaluation," requires the licensee to adopt appropriate procedures to evaluate deviations and failures to comply associated with substantial safety hazards as soon as practicable.

Contrary to the above, a procedure which fulfills the requirements of 10 CFR § 21.21 was not available to the inspection team.

This is a Severity Level V Violation (Supplement VII).

Pursuant to the provisions of 10 CFR § 2.201, DuPont Merck is hereby required to submit a written statement or explanation to the U.S. Regulatory commission, ATTN: Document control Desk, Washington, DC 20555, with a copy to Robert L. Baer, Chief, Source Containment and Devices Branch, Division of Industrial and Medical Nuclear Safety, Office of Nuclear Material Safety and Safeguards (NMSS), within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include, for each violation, (1) the reason for the violation, or, if contested, the basis for disputing the violation; (2) the corrective steps that have been taken and the results achieved; (3) the corrective steps that will be taken to avoid further violations; and (4) the date when full compliance will be achieved.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert Baer

Robert L. Baer, Chief
Source Containment and Devices Branch
Division of Industrial and Medical
Nuclear Safety, NMSS

Dated at Rockville, Maryland
this 23 day of May 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NOTICE OF NONCONFORMANCE

The DuPont Merck Pharmaceutical Company
North Billerica, Massachusetts

Docket No. 07100384

Based on results of the U.S. Nuclear Regulatory Commission inspection conducted on March 30, 1994, it appears that certain of your activities were not conducted in accordance with NRC requirements (refer to Inspection Report No. 94206).

A. 10 CFR § 71.125, "Control of measuring and test equipment," requires the licensee to establish measures to assure that tools used in activities affecting quality are properly controlled, calibrated, and adjusted at specific times to maintain accuracy within necessary limits.

Contrary to the above, the inspection team identified an instance where control of equipment was inadequate.

A torque wrench used to tighten the lid bolts of a packaging for transportation was not calibrated.

B. 10 CFR §71.137, "Audits," requires the licensee to carry out a comprehensive system of planned and periodic audits to verify compliance with all aspects of the Quality Assurance (QA) program and to determine the effectiveness of the program.

Contrary to the above, the inspection team identified instances where audits were inadequate.

1. Documented evidence that internal audits were conducted in 1991 and 1992 was not available to the inspection team.

2. Audit checklists used in the internal audits conducted on October 30 and November 2, 1990, and December 7, 1993, were not available to the inspection team.

3. The 1990 and 1993 internal audits were performed by someone other than the individual authorized by the QA Program.

4. Documentation regarding an assessment of the QA Program in complying with regulatory requirements during the 1990 and 1993 internal audits was not available to the inspection team.

5. Documentation regarding follow-up actions taken, corrective action implementation, and determination of corrective action effectiveness resulting from items identified in the 1990 and 1993 internal audit, was not available to the inspection team.

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PDR ADOCK 07100384
C PDR

Please provide a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555, with a copy to Robert L. Baer, Chief, Source Containment and Devices Branch, Division of Industrial and Medical Nuclear Safety, Office of Nuclear Material Safety and Safeguards (NMSS), within 30 days of the date of the letter transmitting this Notice of Nonconformance. This reply should be clearly marked as a "Reply to a Notice of Nonconformance" and should include, for each Nonconformance: (1) a description of steps that have been or will be taken to correct these items; (2) a description of steps that have been or will be taken to prevent recurrence; and (3) the dates your corrective actions and preventive measures were or will be completed.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert Baer

Robert L. Baer, Chief
Source Containment and Devices Branch
Division of Industrial and Medical
Nuclear Safety, MNSS

Dated at Rockville, Maryland
This 23 day of May, 1994

Duke Power Company
Oconee Nuclear Site
P.O. Box 1439
Seneca, SC 29679

J. W. Hampton
Vice President
(803)885-3499 Office
(803)885-3564 Fax

DUKE POWER

May 17, 1994

U.S. Nuclear Regulatory Commission
Attention Document Control Desk
Washington, DC 20555

Subject: Duke Power Company
Oconee Nuclear Station
Docket No. 50-269, -270
Second Ten Year Inservice Inspection Interval
Request for Relief No. 93-11
Supplementary Information

On November 10, 1993 Duke Power Company submitted the subject Relief from Code Request. In addition to the alternate examinations specified in that submittal Duke proposes to perform a Volumetric Examination on 10 of the welds within the system boundary covered by that request. This will basically be equivalent to the Volumetric examination requirements for class 2 piping welds. The Volumetric examination in conjunction with the operational leakage test provides an acceptable level of quality to assure the structural integrity of this piping in lieu of the code required hydrostatic test.

These examinations will be performed during the Unit 1 EOC 15 and Unit 2 EOC 14 refueling outages. The Unit 1 EOC 15 outage is currently in progress and the Unit 2 EOC 14 outage is scheduled for September 9, 1994.

If there are any questions or further information is needed you may contact D. A. Nix at (803) 885-3634.

Very truly yours,

J W Hampton

J. W. Hampton
Site Vice President

A047
1/0

xc : Mr. L. A. Wiens
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Mr. S. D. Ebneter
Regional Administrator, Region II
U. S. Nuclear Regulatory Commission

Mr. P. E. Harmon
Senior NRC Resident Inspector
Oconee Nuclear Station

Mr. Virgil R. Autry
Bureau of Radiological Health
SC Dept. of Health & Environmental Control
2600 Bull St.
Columbia, SC 29201

Southern California Edison Company

P. O. BOX 128

SAN CLEMENTE, CALIFORNIA 92674-0128

R. W. KRIEGER
VICE PRESIDENT
NUCLEAR GENERATION

TELEPHONE
714-368-6255

May 20, 1994

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

Gentlemen:

Subject: **Docket Nos. 50-361 and 50-362**
Amendment Application Nos. 140 and 124
Changes to Technical Specification 3/4.7.3
"Component Cooling Water System"
San Onofre Nuclear Generating Station
Units 2 and 3

References: 1) January 4, 1990, letter from F. R. Nandy (SCE) to Document Control Desk (NRC), Subject: "Docket Nos. 50-361 and 50-362, Component Cooling Water System, TAC Nos. 71194 and 71195, San Onofre Nuclear Generating Station, Units 2 and 3."

2) December 30, 1993, letter from R. M. Rosenblum (SCE) to Document Control Desk (NRC), Subject: "Proposed Change Number 299, Technical Specification Improvement Project, San Onofre Nuclear Generating Station, Units 2 and 3."

Enclosure 1 contains Amendment Application Nos. 140 and 124 to Facility Operating Licenses NPF-10 and NPF-15 for the San Onofre Nuclear Generating Station (SONGS) Units 2 and 3, respectively. These amendment applications consist of Proposed Change Number (PCN) 387. PCN-387 is a request to revise Units 2 and 3 Technical Specification (TS) 3/4.7.3, "Component Cooling Water System," and the corresponding Bases to support the addition of the Component Cooling Water surge tank Backup Nitrogen Supply (BNS) system.

The proposed revision to TS 3/4.7.3 establishes new operability and surveillance requirements for the BNS system. The proposed Bases of TS 3/4.7.3 document design and operational considerations to support the TS changes. The BNS system was added to the Component Cooling Water system to satisfy commitments made in Reference 1.

By Reference 2, Southern California Edison submitted PCN 299, "Technical Specification Improvement Project (TSIP)." This submittal did not include proposed TSs for the BNS system. Markups of the proposed TSIP pages including the proposed BNS system TSs are included as Enclosure 2.

If you would like additional information regarding this Technical Specification change request, please let me know.

Sincerely,

RWKrieger

R. W. Krieger
Vice President
Nuclear Generation

cc: L. J. Callan, Regional Administrator, NRC Region IV
K. E. Perkins, Jr., Director, Walnut Creek Field Office, NRC Region IV
J. A. Sloan, NRC Senior Resident Inspector, San Onofre Units 1, 2, and 3
M. B. Fields, NRC Project Manager, San Onofre Units 2 and 3
H. Kocol, California Department of Health Services

ENCLOSURE 1

PCN 387

CCW BACKUP NITROGEN SUPPLY SYSTEM

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. for a Class 103 License to Acquire, Possess, and Use a Utilization Facility as Part of Unit No. 2 of the San Onofre Nuclear Generating Station	)))))	DOCKET NO. 50-361 Amendment Application No. 140

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. pursuant to 10 CFR 50.90, hereby submit Amendment Application No. 140

This amendment application consists of Proposed Change Number (PCN) NPF-10-387 to Facility Operating License No. NPF-10. PCN NPF-10-387 is a request to revise San Onofre Unit 2 Technical Specification (TS) 3/4.7.3, "Component Cooling Water System," and the corresponding Bases to TS 3/4.7.3 to support the addition of the Component Cooling Water surge tank Backup Nitrogen Supply system.

Subscribed on this 20th day of May, 1994.

Respectfully submitted,

SOUTHERN CALIFORNIA EDISON COMPANY

By: R.W. Krieger

R. W. Krieger
Vice President
Nuclear Generation

State of California
County of ~~Orange~~ San Diego
On May 20, 1994 before me, Linda L. Rulon, personally appeared R. W. Krieger, personally known to me to be the person whose name is subscribed to the within instrument and acknowledged to me that he executed the same in his authorized capacity, and that by his signature on the instrument the person, or the entity upon behalf of which the person acted, executed the instrument.

WITNESS my hand and official seal.

Signature Linda L. Rulon

LINDA L. RULON
COMM. #958229
Notary Public-California
ORANGE COUNTY
My comm. expires MAR 04,1996

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON	)	
COMPANY, ET AL. for a Class 103 License to	)	DOCKET NO. 50-362
Acquire, Possess, and Use a Utilization	)	
Facility as Part of Unit No. 3 of the	)	Amendment Application No. 124
San Onofre Nuclear Generating Station	)	

SOUTHERN CALIFORNIA EDISON COMPANY ET AL. pursuant to 10 CFR 50.90, hereby submit Amendment Application No. 124

This amendment application consists of Proposed Change Number (PCN) NPF-10-387 to Facility Operating License No. NPF-15. PCN NPF-387 is a request to revise San Onofre Unit 3 Technical Specification (TS) 3/4.7.3, "Component Cooling Water System," and the corresponding Bases to TS 3/4.7.3 to support the addition of the Component Cooling Water surge tank Backup Nitrogen Supply system.

Subscribed on this 20th day of May, 1994.

Respectfully submitted,

SOUTHERN CALIFORNIA EDISON COMPANY

By: RWKrieger

R. W. Krieger
Vice President
Nuclear Generation

State of California
County of ~~Orange~~ San Diego
On May 20, 1994 before me, Linda L. Rulon, personally appeared R. W. Krieger, personally known to me to be the person whose name is subscribed to the within instrument and acknowledged to me that he executed the same in his authorized capacity, and that by his signature on the instrument the person, or the entity upon behalf of which the person acted, executed the instrument.

WITNESS my hand and official seal.

Signature Linda L. Rulon

LINDA L. RULON
COMM. #958229
Notary Public-California
ORANGE COUNTY
My comm. expires MAR 04,1996

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-387

This is a request to revise Technical Specification (TS) 3/4.7.3, "Component Cooling Water System," and the corresponding Bases section for the San Onofre Nuclear Generating Station, Units 2 and 3. The proposed revision of TS 3/4.7.3 will provide operability and surveillance requirements for the Backup Nitrogen Supply (BNS) system. A BNS system is installed for each unit to provide an independent, safety-related, Seismic Category I source of pressurized nitrogen which can be used to maintain component cooling water system surge tank pressure.

EXISTING TECHNICAL SPECIFICATIONS

Attachment A - Unit 2 TS and Bases
Attachment B - Unit 3 TS and Bases

PROPOSED TECHNICAL SPECIFICATIONS

Attachment C - Unit 2 TS and Bases
Attachment D - Unit 3 TS and Bases

DESCRIPTION

Units 2 and 3 Technical Specification (TS) 3/4.7.3, "Component Cooling Water System," provides the Limiting Conditions for Operation (LCO), Actions, and Surveillances for the Component Cooling Water (CCW) system. A Backup Nitrogen Supply (BNS) system was added to both Unit 2 and Unit 3 CCW systems to support CCW operability. This proposed change to Unit 2 and Unit 3 TSs adds operability and surveillance requirements for the BNS system. The proposed Bases for TS 3.7.3 document design and operational considerations which support the TS changes.

The BNS system at each unit is an independent, safety-related, Seismic Category I source of pressurized nitrogen used to maintain the pressure in each CCW critical loop should the normal nitrogen supply become unavailable. The existing Surveillance Requirement (SR) 4.7.3 has demonstrated CCW system operability. The proposed additions to SR 4.7.3 require demonstration of BNS system operability and increase confidence that the CCW system will remain OPERABLE during the applicable operational modes.

The following changes to the Unit 2 and Unit 3 TSs and Bases are proposed:

TS 3/4.7.3

1. The original CCW action statement is identified as ACTION a. A new ACTION statement associated with the BNS system is added as ACTION b.

   ACTION b will be: "With either one or both trains of the Backup Nitrogen Supply (BNS) system inoperable, within 8 hours restore the BNS system train(s) to OPERABLE status or declare the associated CCW loop(s) inoperable."

2. Change the number designation of the original CCW surveillance requirements from 4.7.3 to 4.7.3.1. Add new BNS system surveillance requirements as follows:

   4.7.3.2 The BNS system shall be demonstrated OPERABLE:

   a. At least once per 7 days by verifying that at least nine nitrogen gas bottles are installed with a minimum average bottle pressure of 4232 psig.

   b. At least once per refueling interval by verifying that the third stage pressure regulator of the BNS system is set at 55 psig (± 1.5 psi).

BASES TS 3/4.7.3

1. In the existing paragraph, change "component cooling water" to "Component Cooling Water" and add "(CCW)" in the Bases for both Units 2 and 3. These are editorial changes.

2. The BNS system function and design bases are added.

BACKGROUND

During May and June of 1988, the NRC conducted a special team Safety System Functional Inspection (SSFI) to assess the operational readiness of the San Onofre Units 2 and 3 CCW and Salt Water Cooling (SWC) systems. Following the SSFI, Southern California Edison conducted a program to completely reassess the CCW system.

As a result of this reassessment, it was identified that water hammer in the CCW system could occur during flow transients following certain Design Basis Events. These results were transmitted to the NRC by letter dated December 16, 1988.

By a letter dated January 4, 1990, Edison responded to NRC questions and identified three plant modifications to minimize the potential for CCW system water hammer: 1) provide a qualified BNS system connected to the CCW system surge tanks, 2) relocate throttle valves downstream of the containment emergency air coolers, and 3) optimize surge tank level setpoints. Further engineering analysis of the postulated water hammer was unnecessary with the implementation of these modifications. Items 2 and 3, which do not relate to this proposed TS change, have been completed. The CCW BNS systems were installed in Units 2 and 3 during their respective cycle six refueling outages.

BACKUP NITROGEN SUPPLY (BNS) SYSTEM

FUNCTIONAL OBJECTIVE

Voiding of the CCW system high points may occur during flow transients following depressurization of the CCW surge tank. The formation of voids creates the potential for water hammer as voids collapse during these transients. The non safety-related Auxiliary Gas system nitrogen supply provides normal CCW surge tank pressurization. The BNS system was added to be a safety-related (Seismic

Category I, Quality Class II) source of pressurized nitrogen designed to minimize void formation in the CCW system during emergency conditions.

Installation of the BNS system allows the CCW system to be maintained OPERABLE even if the Auxiliary Gas system is not functional due to a random failure or maintenance. In addition to increasing operational flexibility, the BNS system also ensures that a source of nitrogen is available following Design Basis Events for which Auxiliary Gas system availability cannot be credited.

DESIGN BASIS EVENTS

The BNS system is required following events where both a break develops in the CCW Non-Critical Loop (NCL) and the normal nitrogen supply cannot be credited. This could be caused by either a High Energy Line Break (HELB) inside containment or a Design Basis Earthquake.

A HELB inside the containment is postulated to break a CCW NCL line. The postulated HELBs cover small and large break Loss of Coolant Accidents (LOCAs). A Main Steam Line Break (MSLB) is not postulated due to the augmented inservice inspections performed on the main steam lines inside containment (UFSAR sections 3.6A.2.4.3 and 6.6).

A Design Basis Earthquake could also cause a critical crack in the largest non-Seismic Category I portion of the CCW system NCL. High water outflow occurs from the time the critical crack develops until the surge tank Low-Low level setpoint is reached. As the surge tank water level drops the resultant pressure decreases and actuates the BNS system to maintain CCW system pressure.

BNS SYSTEM DESCRIPTION

The BNS system is designed to support the CCW system. The CCW system is arranged in two independent critical cooling loops and one NCL. Each critical loop has a surge tank, pump, and heat exchanger with full heat removal capacity. A third CCW pump is an installed spare which can be aligned to either critical loop. The NCL is automatically isolated from the rest of the CCW system in response to a Containment Isolation Actuation Signal (CIAS) or a Low-Low surge tank level signal.

During normal plant operation, makeup water is supplied to the surge tank by the Nuclear Service Water system. Nitrogen gas is supplied by the Auxiliary Gas system to the surge tank to assure the CCW piping and equipment are water-solid during normal operation. Neither the Auxiliary Gas system nor the Nuclear Service Water system are safety-related, Seismic Category I and do not support emergency operation functions. A Seismic Category I source of nitrogen is provided to each surge tank by the BNS system at each unit.

(Note: A safety-related CCW makeup water system has been added as a backup to the Nuclear Service Water System to replace the existing Seismic Category I source of water for CCW makeup (fire tankers). Proposed TSs were submitted for the safety-related CCW makeup system by Proposed Change Number (PCN) 418, dated December 30, 1992.)

The BNS system is divided into two independent equivalent trains, with each train supporting one CCW critical loop. Each train includes a bank of ten Department of Transportation (DOT) 3AA 6000 psig nitrogen bottles, installed in dedicated bottle racks and connected by a common discharge header. A pressure gauge, isolation valve, and a check valve are provided for each bottle to prevent bottle pressure equalization and facilitate bottle changeout. The capacity of nine bottles with an average minimum pressure of 4132 psig will maintain system pressure in one CCW critical loop for seven days following a Design Basis Event, before bottle replacement is required.

BNS bottle pressure is reduced to the desired surge tank pressure with four stages of pressure regulation. The first two stages of pressure regulation reduce the nitrogen bottle pressure to approximately 1500 psig (± 100 psi) and 235 psig (± 10 psi), respectively. A pressure relief valve set at 275 psig (± 5 psi) is installed downstream of the second stage regulator to prevent over-pressurization of the downstream piping should the second stage regulator fail open.

The Auxiliary Gas system nitrogen supply line connects to the BNS system piping between the third and fourth stage pressure regulators. A safety-related check valve in the Auxiliary Gas system supply line establishes the BNS system boundary. The third stage BNS pressure regulator is set at 55 psig (± 1.5 psi) which is well below the normal supply pressure in the Auxiliary Gas system nitrogen supply line under the maximum expected flow rate. This setpoint will allow for normal CCW system operation using the Auxiliary Gas system nitrogen supply without actuating the BNS system.

The fourth stage pressure regulator is common for the normal and backup supply paths and is set at 38 psig (± 1 psi). This setpoint assures that the surge tank pressure is at least 27.4 psig while nitrogen is being supplied from the BNS at the highest required flow rate (140 scfm). A pressure relief valve set at 56 psig (± 1.5 psi) is installed downstream of the fourth stage pressure regulator to provide additional protection against surge tank over-pressurization should the fourth stage regulator fail open.

The boundary between the ASME Section III surge tank and non-ASME nitrogen supply piping is established at the normally open isolation valve closest to the tank. This is consistent with Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam and Radioactive-Waste-Containing Components of Nuclear Power Plants."

At an average bottle pressure of 4132 psig, sufficient nitrogen inventory is provided in nine of the ten installed nitrogen bottles to provide seven days of CCW operation without bottle replacement. Within seven days following a Design Basis Event additional nitrogen bottles will be brought from an offsite source to support long-term (120 day) operation.

BNS System Operation

There are three time periods associated with the operation of the BNS system. The first period is for 20.9 seconds immediately following the initiating event. The second period begins when the NCL is isolated, but water makeup to the CCW critical loop(s) is still not available. The third period begins when CCW water makeup is reestablished and continues through long-term operation.

During the first period rapid depressurization of the surge tank occurs as water is lost from the NCL. The BNS system actuates when the nitrogen supply pressure decreases below the BNS system third stage pressure regulator setpoint. The surge tank water level continues to drop until it reaches the Low-Low level setpoint or a CIAS is initiated, at which time the NCL is automatically isolated from the critical loop.

During the second period water makeup to the CCW critical loops is not available. Water makeup to the CCW surge tanks is assumed to be inoperable following a Design Basis Event until either the Seismic Category I fire tankers or the safety-related CCW makeup system is placed in service. The fire tankers are assumed to be unavailable for four hours. The safety-related makeup system is assumed to be unavailable for 30 minutes.

The third period of BNS system operation begins when water makeup is reestablished. During the third period the BNS system supplies nitrogen to replace normal system gas leakage only. No operator action is required for the BNS system for seven days following a Design Basis Event.

DISCUSSION

The San Onofre Units 2 and 3 UFSAR, Section 9.2.2, credits a pressurized CCW surge tank to maintain the CCW system water-solid. During normal operation the surge tank pressure is maintained by the Auxiliary Gas system nitrogen supply. However, the Auxiliary Gas system is not Seismic Category I and is assumed to be non-functional during and after a Design Basis Event. The BNS system was installed as a qualified source of pressurized nitrogen to maintain the CCW surge tank pressure following a Design Basis Event. Therefore, operability of the BNS system must be considered when establishing CCW operability requirements.

Proposed TS 3.7.3 ACTION b establishes the action requirement associated with the BNS system. The proposed Allowable Outage Time (AOT) of 8 hours is based upon the conclusions of a conservative Probabilistic Risk Assessment (PRA).

BNS System Design Bases

The BNS system for each unit consists of two independent, full capacity trains. Each train is dedicated to one CCW surge tank and maintains the redundant CCW system design by providing a safety related, Seismic Category I source of nitrogen for surge tank pressurization.

The BNS system is designed to maintain the surge tank pressure for a minimum of seven days following a Design Basis Event without operator action. A seven-day period of BNS system operation without action is based on reducing the post-LOCA dose rate in the BNS bottle changeout area below 10CFR50, Appendix A, General Design Criterion 19 limits. The seven days of BNS system operation will require approximately 2974 scf of nitrogen which corresponds to a minimum initial average bottle pressure of 4132 psig. The nitrogen inventory is administratively controlled by the proposed TS bottle pressure surveillance. The seven day surveillance requires the average bottle pressure of nine of the ten bottles to be at least 4232 psig, which is 100 psi greater than the operability requirement of 4132 psig. This 100 psi margin accounts for seven days of maximum expected BNS system leakage.

The design capacity of the BNS system currently accounts for a normal CCW system liquid leakage of 6 gpm for a period of four hours following a Design Basis Event, which is the amount of time credited to connect the Seismic Category I fire tankers for makeup water. This BNS design capacity is sufficient to account for a normal CCW system liquid leakage of up to 18 gpm for a period of more than an hour following a Design Basis Event, at which time credit can be taken for the newly installed CCW safety-related makeup water system.

The Auxiliary Gas system nitrogen supply pressure is normally available at 85 to 105 psig. At the supply pressure of 85 psig and a maximum expected nitrogen flow rate during normal operation of 230 scfm the pressure downstream of the third stage regulator will be approximately 59.5 psig. The third stage pressure regulator setpoint of 55 psig (± 1.5 psi) will assure that the BNS system will remain isolated while the Auxiliary Gas system is functional, while being capable of maintaining the surge tank pressure above 27.4 psig should the normal nitrogen supply become unavailable. The BNS system third stage pressure regulator setpoint is administratively controlled by the proposed technical specification surveillance.

The OPERABILITY of the BNS system does not affect normal plant operation because the BNS system is not normally in operation. The BNS system action statements are not normally entered for bottle change out since the BNS system is designed with one more bottle than is required for seven days of BNS operation. The safety function of the BNS system is limited to minimizing void formation in the CCW system under specific emergency circumstances. Therefore, the proposed 8-hour AOT allows sufficient time for normal maintenance and service of the BNS system without unnecessarily shutting the plant down.

Probabilistic Risk Assessment (PRA)

A Probabilistic Risk Assessment (PRA) has been performed to determine the risk of core damage resulting from inoperability of the BNS system. The PRA was intended to qualitatively assess the effect on margin of safety (core damage risk) relative to BNS system inoperability and, consistent with the methodology used for performing PRAs, assign a quantitative core damage risk value. An event tree was developed to model the impact of BNS system unavailability on the risk of core damage following a seismic event with a free-field acceleration between 0.2g and 0.67g. A seismic event with a free-field acceleration greater than 0.67g was not examined because the integrity of the entire CCW system is not assured for such an event.

The risk of losing the Auxiliary Gas system nitrogen supply during a seismic event with a free-field acceleration greater than 0.2 g is assumed to be 1.0 since the system is Quality Class III and Seismic Category II. The PRA model assumed the availability of the Auxiliary Gas system for other events which affect the CCW system.

The consequences of a potential water hammer in a CCW train where the nitrogen pressurization of the associated surge tank is lost has been evaluated, but has not been fully quantified and analyzed in detail. Therefore, it was assumed that loss of nitrogen pressurization in a surge tank leads to failure of the associated train. While this assumption is conservative, the degree of conservatism can not be quantified.

The event tree discussed above was solved to determine the risk of core damage for a base case, where both trains of the BNS system are OPERABLE, and also for the cases where one or both trains of the BNS system are inoperable. The additional risk of core damage due solely to BNS system inoperability was determined by subtracting the base case risk from the results of the cases where one or both trains are considered inoperable. The PRA concluded that the increased risks of core damage with one train of the BNS system inoperable and with both trains of the BNS system inoperable are 8.2E-9 per hour and 9.6E-08 per hour, respectively.

With one train of the BNS system inoperable the proposed TS allows an overall AOT of 80 hours before a plant shutdown is required. This is based on the 8-hour BNS system AOT plus the following 72-hour AOT allowed by the existing TS 3.7.3 for one train of CCW being inoperable. With both trains of the BNS system inoperable, the proposed TS allows an overall AOT of 9 hours, which is based on 8 hours for the BNS inoperable followed by 1 hour allowed by TS 3.0.3 for both trains of CCW inoperable. The overall AOTs of 80 and 9 hours result in core damage risk increases of 6.5E-7 and 8.6E-7, respectively.

Core damage risk for Units 2 and 3 due to internal initiating events is 3.0E-5 per year. Core damage risk due to external events is being evaluated under the Individual Plant Examination of External Events (IPEEE) and is not currently available. The increased risk of core damage due to the BNS system AOTs is less than 3%, which is considered acceptable.

SAFETY ANALYSIS

The proposed change described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The Component Cooling Water (CCW) system removes heat to mitigate the consequences of those design basis accidents included in chapter 15 of the Updated Final Safety Analysis Report (UFSAR). A CCW system failure is not an accident initiating event as listed in the UFSAR, Table 15.0-2. The addition of the Backup Nitrogen Supply (BNS) system does not change the CCW system function and does not interface with any system which relates to the initiating events listed in Table 15.0-2 of the UFSAR. The BNS system is designed to Quality Class II, Seismic Category I requirements and will increase CCW reliability by minimizing CCW system voiding during and after a Design Basis Event (DBE). Failure of the BNS system will not by itself result in an accident or have any effect on normal plant operation.

   The proposed revision of Technical Specification (TS) 3/4.7.3 will not change the CCW system operation. This amendment request retains the original CCW TS requirements and adds provisions specifically limited to the BNS system. The proposed revisions provide an 8-hour Allowed Outage Time (AOT) for one or both trains of the BNS system inoperable to avoid unnecessary plant power reductions. If the 8-hour AOT for BNS system inoperability is not met, the associated CCW train(s) must be declared inoperable. The 8-hour AOT followed by either the 72-hour AOT for one train of CCW inoperable or the 1-hour AOT provided by TS 3.0.3 for both trains of CCW inoperable results in overall AOTs of 80 and 9 hours, respectively. The results of a conservative Probabilistic Risk Assessment demonstrate that for the overall 80-hour and 9-hour AOTs the increases in core damage risk per year are 6.5E-7 and 8.6E-7, respectively. This results in less than a 3% increase in the annual core damage risk for Units 2 and 3.

The proposed revisions to TS 3/4.7.3 include surveillance requirements to provide assurance that the BNS system remains OPERABLE when required to support CCW operation. Therefore, operation of the facility in accordance with this proposed TS change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with this proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: "No"

   The BNS system does not change the CCW system function and does not interface with any system related to the initiating events listed in Table 15.0-2 of the UFSAR. The BNS system is designed to Seismic Category I requirements and will minimize CCW system voiding and the potential for a subsequent water hammer by maintaining the CCW surge tank pressure during and after a DBE. No new High Energy Line Break considerations apply because the nitrogen bottle pressure is reduced at the bottle header and all connections are less than one inch in diameter. The BNS system is independent from all systems possibly related to the initiating DBEs listed in the UFSAR Table 15.0-2.

   The proposed TS 3/4.7.3 revision does not change the existing CCW system requirements. This proposed change adds operability and surveillance requirements for the BNS system to support CCW system operability and provide additional assurance that plant operation is consistent with the design basis. Failure of the BNS system will not by itself result in an accident or have any effect on normal plant operation. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed amendment involve a significant reduction in a margin of safety?

   Response: "No"

   The addition of the BNS system enhanced the CCW system by minimizing the possibility for water hammer following certain postulated events. Surveillance and nitrogen bottle change-out procedures assure that the BNS system is available to perform its safety-related function. The redundant cooling capacity of the CCW system is maintained by providing an independent dedicated BNS system for each CCW critical loop, assuming a single failure.

   The safety function of the BNS system is limited to the minimization of void formation in the CCW system under a specific set of coincident circumstances following a DBE. The proposed revision to TS 3/4.7.3 allows the BNS system to have one or both trains inoperable for 8 hours before the associated CCW train(s) must be declared inoperable. The BNS system AOTs do not affect plant operation because the BNS system is not normally in operation. The BNS system action statements are not normally entered for normal bottle change out since the BNS system is designed with one more bottle than is required for seven days of BNS system operation. Therefore, the proposed changes do not involve a reduction in a margin of safety.

SAFETY AND SIGNIFICANT HAZARDS DETERMINATION

Based on the Safety Analysis, it is concluded that: (1) The proposed change does not constitute a significant hazards consideration as defined by 10CFR50.92 and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change. Moreover, because this action does not involve a significant hazards consideration, it will also not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

ATTACHMENT A

EXISTING TECHNICAL SPECIFICATIONS
AND BASES

UNIT 2

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per refueling interval during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position and each component cooling water pump starts automatically on an SIAS test signal.

3/4.7.1.6 ATMOSPHERIC DUMP VALVES (Continued)

The provisions of Specification 3.0.4 in MODES 2, 3, and 4 do not apply when only one ADV is inoperable, and the ADV can be made OPERABLE within the allowed action times. However, with two inoperable ADVs the plant must be placed on shutdown cooling. Therefore, the provisions of Specification 3.0.4 do apply with two inoperable ADVs.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on secondary side steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 40°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SALT WATER COOLING SYSTEM

The OPERABILITY of the salt water cooling system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

ATTACHMENT B

EXISTING TECHNICAL SPECIFICATIONS
AND BASES

UNIT 3

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per refueling interval during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position and each component cooling water pump starts automatically on an SIAS test signal.

3/4.7.1.6 ATMOSPHERIC DUMP VALVES (Continued)

the ADVs are subject to inservice testing per Surveillance 4.7.1.6.3, the frequency of Surveillance 4.7.1.6.1 is based on the length of a fuel cycle.

The provisions of Specification 3.0.4 in MODES 2, 3, and 4 do not apply when only one ADV is inoperable, and the ADV can be made OPERABLE within the allowed action times. However, with two inoperable ADVs the plant must be placed on shutdown cooling. Therefore, the provisions of Specification 3.0.4 do apply with two inoperable ADVs.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on secondary side steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 90°F and 200 psig are based on a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SALT WATER COOLING SYSTEM

The OPERABILITY of the salt water cooling system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

ATTACHMENT C

PROPOSED TECHNICAL SPECIFICATIONS
AND BASES

UNIT 2

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With either one or both trains of the Backup Nitrogen Supply (BNS) system inoperable, within 8 hours restore the BNS system train(s) to OPERABLE status or declare the associated CCW loop(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per refueling interval during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position and each component cooling water pump starts automatically on an SIAS test signal.

4.7.3.2 The BNS system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that at least nine nitrogen gas bottles are installed with a minimum average bottle pressure of 4232 psig.

b. At least once per refueling interval by verifying that the third stage pressure regulator of the BNS system is set at 55 psig (± 1.5 psi).

3/4.7.1.6 ATMOSPHERIC DUMP VALVES (Continued)

The provisions of Specification 3.0.4 in MODES 2, 3, and 4 do not apply when only one ADV is inoperable, and the ADV can be made OPERABLE within the allowed action times. However, with two inoperable ADVs the plant must be placed on shutdown cooling. Therefore, the provisions of Specification 3.0.4 do apply with two inoperable ADVs.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on secondary side steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 40°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the ~~C~~component ~~C~~cooling ~~W~~water (CCW) system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

The CCW system is normally pressurized to maintain the CCW system water-solid using nitrogen gas supplied to the CCW surge tank by the non-safety related Auxiliary Gas system. Makeup water to the surge tank is normally provided by the non-safety related, Nuclear Service Water system to compensate for normal system leakage.

Following a Design Basis Event, both the non-safety related Auxiliary Gas System and Nuclear Service Water system are assumed to be unavailable. A postulated Design Basis Event could result in CCW system voiding and a subsequent water hammer. The Backup Nitrogen Supply (BNS) system is an independent, safety related, Seismic Category I source of pressurized nitrogen for both CCW surge tanks. The BNS system is designed to minimize CCW system high-point voiding by maintaining the CCW critical loops water-solid during Design Basis Event mitigation.

BNS system OPERABILITY ensures that both CCW surge tanks will be pressurized for at least seven days following a Design Basis Event without bottle changeout. The BNS system is required to be OPERABLE whenever the associated train of CCW is required to be OPERABLE. The BNS system surveillance requirements provide adequate assurance that BNS system OPERABILITY will be maintained.

3/4.7.4 SALT WATER COOLING SYSTEM

The OPERABILITY of the salt water cooling system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

ATTACHMENT D

PROPOSED TECHNICAL SPECIFICATIONS
AND BASES

UNIT 3

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With either one or both trains of the Backup Nitrogen Supply (BNS) system inoperable, within 8 hours restore the BNS system train(s) to OPERABLE status or declare the associated CCW loop(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per refueling interval during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position and each component cooling water pump starts automatically on an SIAS test signal.

4.7.3.2 The BNS system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that at least nine nitrogen gas bottles are installed with a minimum average bottle pressure of 4232 psig.

b. At least once per refueling interval by verifying that the third stage pressure regulator of the BNS system is set at 55 psig (± 1.5 psi).

3/4.7.1.6 ATMOSPHERIC DUMP VALVES (Continued)

the ADVs are subject to inservice testing per Surveillance 4.7.1.6.3, the frequency of Surveillance 4.7.1.6.1 is based on the length of a fuel cycle.

The provisions of Specification 3.0.4 in MODES 2, 3, and 4 do not apply when only one ADV is inoperable, and the ADV can be made OPERABLE within the allowed action times. However, with two inoperable ADVs the plant must be placed on shutdown cooling. Therefore, the provisions of Specification 3.0.4 do apply with two inoperable ADVs.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on secondary side steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 90°F and 200 psig are based on a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Ceomponent Ceooling Wwater (CCW) system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

The CCW system is normally pressurized to maintain the CCW system water-solid using nitrogen gas supplied to the CCW surge tank by the non-safety related Auxiliary Gas System. Makeup water to the surge tank is normally provided by the non-safety related, Nuclear Service Water system to compensate for normal system leakage.

Following a Design Basis Event, both the non-safety related Auxiliary Gas system and Nuclear Service Water system are assumed to be unavailable. A postulated Design Basis Event could result in CCW system voiding and a subsequent water hammer. The Backup Nitrogen Supply (BNS) system is an independent, safety related, Seismic Category I source of pressurized nitrogen for both CCW surge tanks. The BNS system is designed to minimize CCW system high-point voiding by maintaining the CCW critical loops water-solid during Design Basis Event mitigation.

BNS system OPERABILITY ensures that both CCW surge tanks will be pressurized for at least seven days following a Design Basis Event without bottle changeout. The BNS system is required to be OPERABLE whenever the associated train of CCW is required to be OPERABLE. The BNS system surveillance requirements provide adequate assurance that BNS system OPERABILITY will be maintained.

3/4.7.3 SALT WATER COOLING SYSTEM

The OPERABILITY of the salt water cooling system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

ENCLOSURE 2

PCN 387 PROPOSED CHANGES

TECHNICAL SPECIFICATION IMPROVEMENT PROJECT FORMAT

UNITS 2 AND 3

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CCW train inoperable.	A.1	--------NOTE--------- When in MODE 4, enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops – MODE 4" for shutdown cooling made inoperable by CCW. --------------------- Restore CCW train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours
C. Either one or both trains of CCW Backup Nitrogen Supply (BNS) system inoperable.	C.1	Restore inoperable BNS system train(s) to OPERABLE status.	8 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1	Declare associated train(s) of CCW inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	-------------------NOTE------------------- Isolation of CCW flow to individual components does not render the CCW System inoperable. --- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.7.2	Verify each CCW automatic valve in the flow path actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.7.7.3	Perform inservice testing for each CCW manual, power operated, automatic valve, and pump in the flow path servicing safety related equipment.	In accordance with the Inservice Testing Program
SR 3.7.7.4	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.7.7.5	Verify that at least nine nitrogen gas bottles are installed for the BNS system with a minimum average bottle pressure of 4232 psig.	7 days
SR 3.7.7.6	Verify that the third stage pressure regulator of the BNS system is set at 55 psig (+/- 1.5 psi).	24 months

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Salt Water Cooling System, and thus to the environment.

The CCW System is arranged as two independent full capacity cooling loops, and has isolatable nonsafety related components. Each safety related train includes a full capacity pump, surge tank, heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus. A pressurized surge tank in the system ensures sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection actuation signal, and all nonessential components are isolated.

Additional information on the design and operation of the system, along with a list of the components served, is presented in the UFSAR, Section 9.2.2, Reference 1. The principal safety related function of the CCW System is the removal of decay heat from the reactor via the Shutdown Cooling (SDC) System heat exchanger. This may utilize the SCS heat exchanger, during a normal or post accident cooldown and shutdown, or the Containment Spray System during the recirculation phase following a loss of coolant accident (LOCA).

The CCW system is normally pressurized to maintain the CCW system water-solid using nitrogen gas supplied to the CCW surge tank by the non-safety related Auxiliary Gas System. Makeup water to the surge tank is normally provided by the non-safety related Nuclear Service Water system to compensate for normal system leakage.

(continued)

BASES

BACKGROUND (continued)

Following a Design Basis Event, both the non-safety related Auxiliary Gas system and Nuclear Service Water system are assumed to be unavailable. A postulated Design Basis Event could result in CCW system voiding and a subsequent water hammer. The Backup Nitrogen Supply (BNS) system is an independent, safety related, Seismic Category I source of pressurized nitrogen for both CCW surge tanks. The BNS system is designed to minimize CCW system high-point voiding by maintaining the CCW critical loops water-solid during Design Basis Event Mitigation.

BNS system OPERABILITY ensures that both CCW surge tanks will be pressurized for at least seven days following a Design Basis Event without bottle changeout. The BNS system is required to be OPERABLE whenever the associated train of CCW is required to be OPERABLE. The BNS system surveillance requirements provide adequate assurance that BNS system OPERABILITY will be maintained.

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is for one CCW train in conjunction with a 100% capacity Containment Cooling System (containment spray, containment coolers, or a combination) removing core decay heat 20 minutes after a design basis LOCA. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System (RCS) by the safety injection pumps.

The CCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

The CCW System also functions to cool the unit from SDC entry conditions (T_{cold} < 350°F) to MODE 5 (T_{cold} < 200°F) during normal and post accident operations. The time required to cool from 350°F to 200°F is a function of the number of CCW and SDC trains operating. One CCW train is sufficient to remove decay heat during subsequent operations with T_{cold} < 200°F. This assumes that a maximum seawater temperature of 76°F occurs simultaneously with the maximum heat loads on the system.

The CCW System satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES

LCO

The CCW trains are independent of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a DBA, one CCW train is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two CCW trains must be OPERABLE. At least one CCW train will operate assuming the worst single active failure occurs coincident with the loss of offsite power.

A CCW train is considered OPERABLE when the following:

a. The associated pump and surge tank are OPERABLE; and

b. The associated piping, valves, heat exchanger and instrumentation and controls required to perform the safety related function are OPERABLE; and

c. The associated train of the BNS system is OPERABLE.

The isolation of CCW from other components or systems not required for safety may render those components or systems inoperable, but does not affect the OPERABILITY of the CCW System.

(continued)

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CCW train inoperable.	A.1	--------NOTE--------- When in MODE 4, enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops – MODE 4" for shutdown cooling made inoperable by CCW. --------------------- Restore CCW train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours
C. Either one or both trains of CCW Backup Nitrogen Supply (BNS) system inoperable.	C.1	Restore inoperable BNS system train(s) to OPERABLE status.	8 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1	Declare associated train(s) of CCW inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	-------------------NOTE------------------- Isolation of CCW flow to individual components does not render the CCW System inoperable. --- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.7.2	Verify each CCW automatic valve in the flow path actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.7.7.3	Perform inservice testing for each CCW manual, power operated, automatic valve, and pump in the flow path servicing safety related equipment.	In accordance with the Inservice Testing Program
SR 3.7.7.4	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	24 months
SR 3.7.7.5	Verify that at least nine nitrogen gas bottles are installed for the BNS system with a minimum average bottle pressure of 4232 psig.	7 days
SR 3.7.7.6	Verify that the third stage pressure regulator of the BNS system is set at 55 psig (+/- 1.5 psi).	24 months

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Salt Water Cooling System, and thus to the environment.

The CCW System is arranged as two independent full capacity cooling loops, and has isolatable nonsafety related components. Each safety related train includes a full capacity pump, surge tank, heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus. A pressurized surge tank in the system ensures sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection actuation signal, and all nonessential components are isolated.

Additional information on the design and operation of the system, along with a list of the components served, is presented in the UFSAR, Section 9.2.2, Reference 1. The principal safety related function of the CCW System is the removal of decay heat from the reactor via the Shutdown Cooling (SDC) System heat exchanger. This may utilize the SCS heat exchanger, during a normal or post accident cooldown and shutdown, or the Containment Spray System during the recirculation phase following a loss of coolant accident (LOCA).

The CCW system is normally pressurized to maintain the CCW system water-solid using nitrogen gas supplied to the CCW surge tank by the non-safety related Auxiliary Gas System. Makeup water to the surge tank is normally provided by the non-safety related Nuclear Service Water system to compensate for normal system leakage.

(continued)

BASES

BACKGROUND (continued)

Following a Design Basis Event, both the non-safety related Auxiliary Gas system and Nuclear Service Water system are assumed to be unavailable. A postulated Design Basis Event could result in CCW system voiding and a subsequent water hammer. The Backup Nitrogen Supply (BNS) system is an independent, safety related, Seismic Category I source of pressurized nitrogen for both CCW surge tanks. The BNS system is designed to minimize CCW system high-point voiding by maintaining the CCW critical loops water-solid during Design Basis Event Mitigation.

BNS system OPERABILITY ensures that both CCW surge tanks will be pressurized for at least seven days following a Design Basis Event without bottle changeout. The BNS system is required to be OPERABLE whenever the associated train of CCW is required to be OPERABLE. The BNS system surveillance requirements provide adequate assurance that BNS system OPERABILITY will be maintained.

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is for one CCW train in conjunction with a 100% capacity Containment Cooling System (containment spray, containment coolers, or a combination) removing core decay heat 20 minutes after a design basis LOCA. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System (RCS) by the safety injection pumps.

The CCW System is designed to perform its function with a single failure of any active component, assuming a loss of offsite power.

The CCW System also functions to cool the unit from SDC entry conditions (T_{cold} < 350°F) to MODE 5 (T_{cold} < 200°F) during normal and post accident operations. The time required to cool from 350°F to 200°F is a function of the number of CCW and SDC trains operating. One CCW train is sufficient to remove decay heat during subsequent operations with T_{cold} < 200°F. This assumes that a maximum seawater temperature of 76°F occurs simultaneously with the maximum heat loads on the system.

The CCW System satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES

LCO

The CCW trains are independent of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a DBA, one CCW train is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two CCW trains must be OPERABLE. At least one CCW train will operate assuming the worst single active failure occurs coincident with the loss of offsite power.

A CCW train is considered OPERABLE when the following:

a. The associated pump and surge tank are OPERABLE; and

b. The associated piping, valves, heat exchanger and instrumentation and controls required to perform the safety related function are OPERABLE; and

c. The associated train of the BNS system is OPERABLE.

The isolation of CCW from other components or systems not required for safety may render those components or systems inoperable, but does not affect the OPERABILITY of the CCW System.

(continued)

ROUTING AND TRANSMITTAL SLIP

Date 5-18-94

TO: (Name, office symbol, room number, building, Agency/Post)	Initials	Date
DCS P1 37		
2. PDR		
3.		
4.		
5.		

Action		File		Note and Return
Approval		For Clearance		Per Conversation
As Requested		For Correction		Prepare Reply
Circulate		For Your Information		See Me
Comment		Investigate		Signature
Coordination		Justify		

REMARKS

This previous Central File material can now be made publicly available.

MATERIAL RELATED TO CRGR MEETING NO. 197

CC (LIST ONLY) JEAN RATAJE, PDR L STREET

DO NOT use this form as a RECORD of approvals, concurrences, disposals, clearances, and similar actions

FROM: (Name, org. symbol, Agency/Post)	Room No.—Bldg.
DENNIS ALLISON	Phone No. 24148

5041-102

OPTIONAL FORM 41 (Rev. 7-76)
Prescribed by GSA
FPMR (41 CFR) 101-11.206

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PDR REVGP NRGCRGR
MEETING197 PDR

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SENT TO PDR ON ________

MATERIAL RELATED TO CRGR MEETING NO. 197
TO BE MADE PUBLICLY AVAILABLE

1. MEMO FOR J. TAYLOR FROM E. JORDAN DATED 1-10-91
SUBJECT: MINUTES OF CRGR MEETING NUMBER 197
INCLUDING THE FOLLOWING ENCLOSURES WHICH WERE NOT PREVIOUSLY RELEASED:

a. ENCLOSURE 2
A SUMMARY OF DISCUSSIONS OF A PROPOSED Generic Letter on Licensee Commercial-Grade Dedication and Procurement Programs

b. ENCLOSURE ________
A SUMMARY OF DISCUSSIONS OF A PROPOSED

c. ENCLOSURE ________
A SUMMARY OF DISCUSSIONS OF A PROPOSED

2. MEMO FOR E. JORDAN FROM F. Miraglia DATED 11-28-90
FORWARDING REVIEW MATERIALS ON A PROPOSED GL on Licensee Commercial-Grade Procurement + Dedication Programs

3. MEMO FOR E JORDAN FROM ________ DATED ________
FORWARDING REVIEW MATERIALS ON A PROPOSED

4. MEMO FOR E JORDAN FROM ________ DATED ________
FORWARDING REVIEW MATERIALS ON A PROPOSED

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

NOV 28 1990

MEMORANDUM FOR: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

FROM: Frank J. Miraglia, Deputy Director
Office of Nuclear Reactor Regulation

SUBJECT: PROPOSED GENERIC LETTER ON LICENSEE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS

The Office of Nuclear Reactor Regulation requests that the Committee to Review Generic Requirements (CRGR) consider the enclosed proposed generic letter. The staff is proposing the enclosed generic letter to notify the industry of the staff's intent to pause in conducting programmatic procurement inspection and enforcement activities and to identify a number of failures in the licensees' commercial-grade dedication programs identified during past NRC inspections. This generic letter also provides information from the NRC's inspections of the licensees' commercial-grade procurement and dedication programs which, if included in licensees' implementation of these programs, could have avoided violations of regulatory requirements.

The commercial-grade dedication inspection findings discussed in Enclosure 1 to the generic letter are based upon 10 CFR Part 50, Appendix B requirements and do not involve changes in the staff's positions. Further, the proposed generic letter does not require any specific licensee action or response to the NRC based on the issuance of this generic letter. Because no new regulations or regulatory practices are involved, the relation to the Commission's safety goals have not been explicitly addressed. However, this action appears to relate to how well a plant is operated. The matters addressed in this generic letter contribute to reducing or avoiding a substantial increase in uncertainty in the assumptions on which safety goal calculations are based.

Enclosure 2 to this memorandum is the proposed generic letter and Enclosure 1 contains the CRGR review package. Brian K. Grimes, Director, Division of Reactor Inspection and Safeguards, is the sponsoring division director. OGC concurrence is currently being sought.

Frank J. Miraglia

Frank J. Miraglia, Deputy Director
Office of Nuclear Reactor Regulation

Enclosure:
1. CRGR Review Package
2. Draft Generic Letter on Licensee Commercial-Grade Procurement and Dedication Programs

CONTACT: Richard P. McIntyre, NRR
492-3215

~~9012100328~~ 14pp

ENCLOSURE 1

CRGR REVIEW PACKAGE

RESPONSE TO REQUIREMENTS FOR CONTENT OF PACKAGE SUBMITTED FOR CRGR REVIEW

(i) The proposed generic requirement as it is proposed to be sent out to all holders of operating licenses and construction permits for nuclear power plants.

The staff position is:

The proposed position is stated in the proposed generic letter. In summary, all holders of operating licenses and construction permits for nuclear power reactors would be notified of the staff's intent to pause in conducting programmatic procurement inspection and enforcement activities. However, the NRC will conduct selected assessments to determine the progress of the industry in improving procurement and dedication programs. (Utilities are now implementating the Nuclear Management Resources Council (NUMARC) Initiative on the Dedication of Commercial-Grade Items and the Comprehensive Procurement Initiative). This generic letter identifies a number of failures in the licensees' commercial-grade dedication programs that were identified during recent NRC inspections. In addition, this generic letter provides the staff's views on key activities, which, if included in licensee implementation of these programs, could have avoided such failures.

(ii) Draft staff papers or other underlying staff documents supporting the requirements or staff (regulatory) positions. (A copy of all materials referenced in the document shall be made available upon request to the CRGR staff. Any committee member may request that the CRGR staff obtain a copy of any referenced material for his or her use.)

The following documents support the staff's position:

(a) Proposed NRC Generic Letter 90-XX: "Licensee Commerical-Grade Procurement and Dedication Programs" (See generic letter in Enclosure 2).

(b) Enclosure 1 of the proposed generic letter lists 13 NRC Inspection Reports regarding licensees' procurement and dedication programs.

(c) NRC Generic Letter 89-02: "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products." Enclosure 1 to Generic Letter 89-02 lists NRC bulletins and information notices regarding nonconforming materials and equipment.

(d) SECY-90-057, Advance Notice of Proposed Rulemaking, "Acceptance of Products Purchased for Use in Nuclear Power Plant Structures, Systems, and Components."

(e) SECY-90-304, "NUMARC Initiatives on Procurement."

(f) SECY-90-261, "Inspection and Enforcement Initiatives for Commercial-Grade Procurement and Dedication Programs."

(iii) Each proposed requirement or staff (regulatory) position shall contain the sponsoring office's position as to whether the proposal would increase requirements or staff (regulatory) positions, implement existing requirements or staff (regulatory) positions, or would relax or reduce existing requirements or staff (regulatory) positions.

The commercial-grade dedication approaches discussed in Enclosure 1 of the proposed generic letter do not constitute new NRC requirements or positions, but provide specific clarifications to implementation guidance to meet 10 CFR Part 50, Appendix B. However, if current or improved procurement activities identify shortcomings in the form, fit, or function of specific vendor products or if failure experience or current information on supplier adequacy indicates that a component may not be suitable for service, corrective actions should include a look-back for all such installed and stored items. The licensees' actions in this regard for both warehouse and installed items should follow the existing requirements for corrective action and follow-up contained in Criterion XVI of 10 CFR Part 50, Appendix B.

(iv) The proposed method of implementation along with the concurrence (and any comments) of OGC on the method proposed.

The staff proposes to promulgate the clarification by means of a generic letter. This method has been effective in the past. The Office of the General Counsel (OGC) has provided comments and has concurred in the proposed generic letter.

(v) Regulatory analyses generally conforming to the directives and guidance of NUREG/BR-0058 and NUREG/CR-3568. (Make sufficient to address the Paperwork Reduction Act, the Regulatory Flexibility Act, and Executive Order 12291).

(a) This request for information was approved by the Office of Management and Budget under blanket clearance number 3150-0011 as meeting the requirements of the Paper Reduction Act and Executive Order 12291.

(b) Because this request is not a rulemaking action, the Regulatory Flexibility Act does not apply.

(vi) Identification of the category of reactor plants to which the generic requirement or staff position is to apply (that is, whether it is to apply to new plants only, new OLs only, OLs after a certain date, OLs before a certain date, all OLs, all plants under construction, all plants, all water reactors, all PWRs only, some vendor types, some vintage types such as BWR 6 and 4, jet pump and nonjet pump plants, etc.)

As described in Item (i) above, the proposed requirements apply to all holders of operating licenses and construction permits for nuclear power reactors.

(vii) For each such category of reactor plants, an evaluation which demonstrates how the action should be prioritized and scheduled in light of other ongoing regulatory activities. The evaluation shall document for consideration information available concerning any of the proposed backfit factors as may be appropriate and any other information relevant and material to the proposed action:

Response to this item is not required pursuant to Revision 4 of the CRGR Charter, Section III.D., because the proposed generic letter announces an NRC inspection pause and conforming to the staff views on key dedication activities would bring licensees into compliance with existing regulatory requirements. This action should not affect the industry's schedule for improvements because the initiative on commerical-grade dedication was implemented in early 1990 and the comprehensive procurement initiative is already underway.

(viii) For each evaluation conducted pursuant to 10 CFR 50.109, the proposing office director's determination, together with the rationale for the determination based on the considerations of paragraphs (i) through (vii) above, that:

(a) There is a substantial increase in the overall protection of public health and safety or the common defense and security to be derived from the proposal; and

This item is not applicable since no changes in staff positions are involved. However, the following discussion provides the safety significance of this action:

The NRC has identified numerous instances in which the nuclear industry received, accepted, and installed products that were not of the quality identified by the manufacturer or supplier. The NRC has also identified examples of significant deficiencies in the procurement and dedication of commercial-grade items, with errors traceable to both suppliers and purchasers who dedicate the items for safety-related applications.

The inadequate dedication of commercial-grade items by suppliers and purchasers (including licensees), increases the probability that hardware installed in safety-related applications may not perform as desired. Therefr e, the guidance in the proposed generic letter provides for overa.l protection of public health and safety.

The NUMARC Initiative on the Dedication of Commercial-Grade Items requested that utilities review and, if necessary, develop or upgrade current programs to meet the intent of Electric Power Research Institute (EPRI) NP-5652. Generic Letter 89-02 conditionally endorses EPRI NP-5652 as a guideline for commercial-grade dedication. The EPRI guideline presents several approaches to implement existing requirements as they apply to commercial-grade items.

(b) The direct and indirect costs of implementation, for the facilities affected, are justified in view of this increased protection.

(1) Direct and indirect costs associated with the required actions by the generic letter result primarily from the evaluation by licensees of their existing procurement programs, and, for deficient programs, the necessary corrective actions. The licensees are performing this review as a result of the NUMARC initiative and should not require substantial additional resources in order to consider the staff views expressed in the generic letter.

The amount of effort needed to correct deficient programs will be a function of the current adequacy of licensee's programs and may range from no changes to changes that require several FTEs each year. The staff believes that the costs of implementation are justified in view of the need to ensure the suitability of materials and equipment procured for use in nuclear safety-related applications.

(2) Occupational radiation exposure should not increase because of the actions requested by this generic letter.

(3) NRC resources will be required to conduct selected assessments to determine the progress of the industry in implementation of the initiative on the dedication of commercial-grade items.

(ix) For each evaluation conducted for proposed relaxations or decreases in current requirements of staff positions, the action is justified because of the proposing office director's determination, together with the rationale for the determination based on the considerations of the above, that:

(a) the public health and safety and the common defense and security would be adequately protected if the proposed reduction in requirements or (regulatory) positions were implemented; and

(b) the cost savings attributed to the action would be substantial enough to justify taking the action.

This item is not applicable to the proposed generic letter because the staff is not proposing a relaxation or decrease in current requirements.

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

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ENCLOSURE 2

TO: ALL HOLDERS OF OPERATING LICENSES AND CONSTRUCTION PERMITS FOR NUCLEAR POWER REACTORS

SUBJECT: LICENSEE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS (GL 90-XX)

This generic letter notifies the industry of the staff's intent to pause in conducting certain procurement inspection and enforcement activities and to identify a number of failures in the licensees' commercial-grade dedication programs identified during recent inspections performed by the U.S. Nuclear Regulatory Commission (NRC). This generic letter also provides further discussion of the staff's views on key activities which, if included in licensee implementation of these programs, could have avoided such failures.

During 1986 to 1989, the NRC has conducted inspections of the licensees' procurement and commercial-grade dedication programs. During these inspections, the NRC staff identified a common, programmatic deficiency in the licensees' control of the procurement and dedication of commercial-grade items for safety-related applications. In a number of cases, the staff found that licensees had not maintained programs to ensure the suitability of equipment for safety-related applications. In addition, the staff identified equipment of indeterminate quality installed in the licensee's facilities.

The NRC staff believes that these inspection findings, in part, indicate a change in the industry's procurement practices and the decrease in the number of qualified nuclear-grade vendors. Ten years ago, licensees made most procurements for major assemblies from approved vendors with programs pursuant to Appendix B of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR). Currently, licensees are increasing the numbers of commercial-grade replacement parts that they procure for use in safety-related applications. This has resulted in a shift of responsibility for ensuring the quality of the item purchased from the suppliers to the licensees. Therefore, dedication processes for commercial-grade parts have increased in importance and NRC inspections have determined that a number of licensees have not satisfactorily performed this dedication process.

The industry should be fully aware of the NRC's concerns in this program area. In the past, escalated enforcement cases have provided notice to the affected licensees and to the industry of NRC's findings, concerns, and expectations in the implementation of procurement and dedication programs. Further, the NRC staff continues to participate in numerous industry meetings and conferences to discuss the NRC's positions in this area.

The Nuclear Utility Management and Resources Council (NUMARC) Board of Directors recently approved a comprehensive procurement initiative which if effectively implemented should markedly improve the assurance that quality components are installed in nuclear power plants. While monitoring industry implementation of these programs, the NRC staff is deferring inspections of licensees' procurement and commercial-grade dedication processes for about a year to allow utilities to have sufficient time to fully understand and implement the guidance being developed by the industry.

However, the NRC will continue to perform certain types of inspection activities. For example, the staff will conduct selected assessments to determine the progress of the industry in improving the procurement and dedication processes. The staff will continue to perform reactive inspections relating to operational events or to defective equipment and, as required, will continue to initiate resultant enforcement actions which will not be affected by the decision to defer programmatic inspections. In addition, the staff will continue to perform inspections of vendors. To further encourage timely and effective implementation of the NUMARC initiatives, the staff will not initiate enforcement action in cases of past programmatic violations that have been adequately corrected. In addition, the staff does not expect licensees to review all past procurements. However, if during current procurement activities, licensees identify shortcomings in the form, fit, or function of specific vendor products, or if failure experience or current information on supplier adequacy indicates that a component may not be suitable for service, corrective actions should include a look-back for all such installed and stored items. In performing these actions for both stored and installed items, licensees should follow the existing requirements for corrective and follow-up actions contained in Criterion XVI of 10 CFR Part 50, Appendix B. A licensee should determine programmatic root causes when actual deficiencies in several different vendor products are identified during current procurement activities and when these deficiencies lead to the replacement of installed or warehouse items as part of corrective action. In such cases, a further sampling of previously procured commercial-grade items may be warranted.

NRC Generic Letter 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," described the NRC's observations on good practices in procurement and provided the NRC's conditional endorsement of an industry standard (EPRI NP-5652) on methods of commercial-grade procurement and dedication. A number of inspection findings indicate that licensees have failed to include certain key activities as appropriate in the implementation of the dedication process. Enclosure 1 includes further discussion of the NRC staff's views on the successful implementation of licensees' programs for commercial-grade dedication. The commercial-grade dedication approaches discussed in Enclosure 1 do not constitute new NRC requirements or positions. We will continue to meet with the industry to ensure a common understanding of implementation issues in this area.

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Although no response to this letter is required, if you have any questions regarding this matter, please contact the persons listed below.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Characteristics of Effective Commercial-Grade Procurement and Dedication Programs
2. List of Recently Issued Generic Letters

Technical Contact: Richard P. McIntyre, NRR
(301) 492-3215

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Enclosure 1

CHARACTERISTICS OF EFFECTIVE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS

Background

Appendix B to 10 CFR Part 50 contains the NRC's regulations for procurement quality assurance (QA) and quality control (QC) for products to be used in safety-related applications. In addition, the NRC has provided further guidance in Regulatory Guides 1.28, 1.33, and 1.123. These requirements and guides assure the suitability of equipment, including commercial-grade items for use in safety-related systems. Criterion III of Appendix B requires licensees to select and review for suitability of application materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems, and components. Criterion IV requires that procurement documents specify the applicable requirements necessary to ensure functional performance. Criterion VII requires licensees to assure that the following are sufficient to identify whether specification requirements for the purchased material and equipment have been met: source evaluation and selection, objective evidence of quality, inspection of the source, and examination of products upon delivery. The process used to satisfy these requirements when upgrading commercial-grade items for safety-related applications is commonly called "dedication." The process of ensuring compliance with 10 CFR Part 50, Appendix B, must include all those activities necessary to establish and confirm the quality and suitability of those items to be installed in safety-related applications. Some of the dedication activities may occur early in the procurement cycle, before the item is accepted from the manufacturer. (10 CFR Part 21 has a more restricted definition of commercial-grade item dedication related to responsibility for evaluation and reporting of defects.) Generic Letter (GL) 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," discussed commercial-grade dedication in terms of engineering involvement in the procurement process, product acceptance, and the dedication process as identified in the EPRI NP-5652 guidelines. This enclosure provides examples of specific failures by licensees to fully implement certain key activities for dedicating and ensuring the suitability of commercial-grade products for safety-related applications. Appropriate implementation of these key activities would have avoided the failures in procurement and commercial-grade dedication observed during past NRC inspections.

Inspection Observations and Findings

From 1986 to 1989, headquarters and regional personnel conducted 13 team inspections of licensees' procurement and dedication programs. These inspections have identified a common, broad programmatic deficiency in licensees' control over the procurement and dedication of commercial-grade items. In a number of cases, licensees have not maintained programs to ensure

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the suitability of equipment for use in safety-related applications as required by 10 CFR Part 50, Appendix B, Criterion III. From these 13 inspections, the staff identified 8 findings that were considered to be Severity Level III violations and 3 findings that were Severity Level IV violations. At one plant, the staff did not assign a severity level to individual violations. Instead, the staff considered the entire group to be a Severity Level III problem and used enforcement discretion, as provided under the shutdown policy, based on the licensee's corrective actions (see 10 CFR Part 2, Appendix C, Section V.G.2). Only one of the plants that were inspected did not receive violations in this program area.

In GL 89-02, the NRC has conditionally endorsed the dedication methods described in EPRI NP-5652 guidelines. The staff believes that licensees who implement these dedications methods, in accordance with the NRC's endorsement, can establish a basis for satisfying the existing requirements of Appendix B to 10 CFR Part 50 as these requirements apply to the dedication process for commercial-grade items. An effective commercial-grade dedication program should include provisions to demonstrate that a dedicated item is suitable for safety-related applications. For a licensee to adequately establish suitability, certain key activities must be performed as appropriate as part of the dedication process.

During each of the 13 inspections, the staff identified a common element in each of the inspection findings. This element was the failure of the licensee to assure that a commercially procured and dedicated item was suitable for the intended safety-related application. In its ability to perform its intended safety function, a dedicated commercial-grade item should be equivalent to the same item procured under a 10 CFR Part 50, Appendix B QA program. The following is a list of the 13 licensees inspected and the inspection report numbers. A summary of the general inspection findings and NRC observations on these findings follows the list of licensee inspections.

LICENSEE and PLANT	INSPECTION REPORT NO.
1. Tennessee Valley Authority (Sequoyah)	50-327/86-61 50-328/86-61
2. Southern California Edison (San Onofre)	50-206/87-02 50-361/87-03 50-362/87-04
3. Alabama Power (Farley)	50-348/87-11 50-364/87-11
4. Louisiana Power and Light (Waterford)	50-382/87-19

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LICENSEE and PLANT		INSPECTION REPORT NO.
5.	Sacramento Municipal Utility District (Rancho Seco)	50-312/88-02
6.	Maine Yankee Atomic Power (Maine Yankee)	50-309/88-200
7.	Northern States Power (Prairie Island)	50-282/88-201 50-306/88-201
8.	Portland General Electric (Trojan)	50-344/88-39 50-344/88-46
9.	Connecticut Yankee Atomic Power (Haddam Neck)	50-213/89-200
10.	Washington Public Power Supply System (WNP-2)	50-397/89-21 50-397/89-28
11.	Florida Power (Crystal River)	50-302/89-200
12.	Gulf States Utilities (River Bend)	50-458/89-200
13.	Commonwealth Edison (Zion)	50-295/89-200 50-304/89-200

1. Inspection Findings

   a. Failure to identify the methods and acceptance criteria for verifying the critical characteristics, such as during receipt inspection, dedication process, or post-installation testing.

   b. Failure to establish verifiable, documented traceability of complex commercial-grade items to their original equipment manufacturers in those cases where the dedication program cannot verify the critical characteristics.

   c. Failure to recognize that some commercial-grade items cannot be fully dedicated once received on site. Certain items are manufactured using special processes, such as welding and heat treating. Dedication testing of these items as finished products would destroy them. For these items, licensees may need to conduct vendor surveillances or to witness certain activities during the manufacturing process.

Discussion

The NRC staff has met on several occasions with NUMARC and licensee representatives to discuss "critical characteristics" as used in the context of commercial-grade procurement and dedication. The term "critical

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characteristics" is not contained in Appendix B and has no special regulatory significance beyond its use and definition in various industry guides and standards. The NRC has not taken the position that all design requirements must be considered to be critical characteristics as defined and used in EPRI NP-5652. Rather, as stated in Appendix B, Criterion III, licensees must assure the suitability of all parts, materials, and services for their intended safety-related applications (i.e., there needs to be assurance that the item will perform its intended safety function when required). The licensee is responsible for identifying the important design, material, and performance characteristics for each part, material, and service intended for safety-related applications, establishing acceptance criteria, and providing reasonable assurance of the conformance of items to these criteria. There is no minimum or maximum number of critical characteristics that need to be verified. Further, the critical characteristics for an item may vary from application to application depending on the design and performance requirements unique to each application.

A licensee may take different approaches for the verification of the critical characteristics, depending on the complexity of the item. In many cases, the licensee can verify the critical characteristics of a simple item during the receipt inspection. However, for a complex item with internal parts which receive special processing during manufacturing, the licensee would probably need to audit or survey the vendor to verify the critical characteristics necessary for the item to perform its safety function. When the dedication program cannot verify the critical characteristics, the licensee should establish documented, verifiable traceability to the original equipment manufacturer. For simple items with critical characteristics that can be verified for the most severe or limiting plant application, the licensee might prefer a broad dedication program to identify and verify the item's critical characteristics to qualify that item for all possible plant applications. For complex items that would be purchased for specific plant applications, the licensee should address the acceptance criteria for each item individually. Engineering involvement is essential in either method because the technical evaluation will identify the critical characteristics, acceptance criteria, and the methods to be used for verification.

2. Inspection Findings

   a. Failure to demonstrate that a like-for-like replacement item is identical in form, fit, and function to the item it is replacing. Part number verification is not sufficient because of the probability of undocumented changes in the design, material, or fabrication of commercial-grade items using the same part number.

   b. Failure to evaluate changes in the design, material, or manufacturing process for the effect of these changes on safety function performance (particularly under design basis event conditions) of replacement items that are similar as opposed to identical to the items being replaced.

c. Failure to ensure that items will function under all design requirements. On some occasions, licensees only ensured that the commercial-grade item would function under normal operation conditions.

d. Failure to verify the validity of certificates of conformance received from vendors not on the licensee's list of approved vendors/suppliers. An unverified certificate of conformance from a commercial-grade vendor is not sufficient.

Discussion

A like-for-like replacement is defined as the replacement of an item with an item that is identical. A like-for-like replacement does not change the engineering analysis or as-built configuration of the component or system in which it is installed, and the replacement item meets the same design specifications, technical and quality requirements, and functional characteristics as the item it replaces. If differences from the original item are identified in the replacement item, then the item is not identical, but similar to the item being replaced, and an evaluation must be performed to determine if any changes in design, material, or the manufacturing process could impact the functional characteristics and ultimately the component's ability to perform its required safety function.

If the licensee can demonstrate that the replacement item is identical, then the licensee need not identify the safety function or review and verify the design requirements and critical characteristics. For example, the replacement item would be identical if it was purchased at the same time from the same vendor as the item it is replacing, or if the user can verify that there have been no changes in the design, materials, or manufacturing process since procurement of the item being replaced.

Engineering involvement is essential in the above activities. The extent of this involvement is dependent on the nature, complexity, and use of the items to be dedicated. Engineering personnel should participate in the procurement process, and product acceptance, to develop purchase specifications, determine specific testing requirements applicable to the products, and evaluate the test results. When engineering personnel specify design requirements for inclusion on the purchase documents for replacement components, they need not reconstruct and reverify the design adequacy, but only ensure that these design requirements (which may reference the original design basis) are properly translated into the purchase order.

Reliance on part number verification and certification documentation is insufficient to ensure the quality of commercially procured products. To conduct effective product acceptance programs, licensees should ensure that these programs include receipt and source inspection, appropriate testing criteria, effective vendor audits (including witness/hold points), special tests and inspections, and post-installation tests. The licensees should establish procedures to implement their programs and should ensure that the implementing personnel have adequate qualifications and training.

DRAFT

MEMORANDUM FOR: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

NOV 2 8 1990

FROM: Frank J. Miraglia, Deputy Director
Office of Nuclear Reactor Regulation

SUBJECT: PROPOSED GENERIC LETTER ON LICENSEE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS

The Office of Nuclear Reactor Regulation requests that the Committee to Review Generic Requirements (CRGR) consider the enclosed proposed generic letter. The staff is proposing the enclosed generic letter to notify the industry of the staff's intent to pause in conducting programmatic procurement inspection and enforcement activities and to identify a number of failures in the licensees' commercial-grade dedication programs identified during past NRC inspections. This generic letter also provides information from the NRC's inspections of the licensees' commercial-grade procurement and dedication programs which, if included in licensees' implementation of these programs, could have avoided violations of regulatory requirements.

The commercial-grade dedication inspection findings discussed in Enclosure 1 to the generic letter are based upon 10 CFR Part 50, Appendix B requirements and do not involve changes in the staff's positions. Further, the proposed generic letter does not require any specific licensee action or response to the NRC based on the issuance of this generic letter. Because no new regulations or regulatory practices are involved, the relation to the Commission's safety goals have not been explicitly addressed. However, this action appears to relate to how well a plant is operated. The matters addressed in this generic letter contribute to reducing or avoiding a substantial increase in uncertainty in the assumptions on which safety goal calculations are based.

Enclosure 2 to this memorandum is the proposed generic letter and Enclosure 1 contains the CRGR review package. Brian K. Grimes, Director, Division of Reactor Inspection and Safeguards, is the sponsoring division director. OGC concurrence is currently being sought.

Original signed by
Frank J. Miraglia
Frank J. Miraglia, Deputy Director
Office of Nuclear Reactor Regulation

OFC	:VIB:DRIS:NRR	:VIB:DRIS:NRR	:DD:DRIS:NRR	:D:DRIS:NRR	:		
NAME	:RPMcIntyre:mkm*	:UPotapovs*	:BDLiaw	:BKGrimes BKG	:		
DATE	:10/25/90	:10/25/90	:11/ /90	:11/21/90	:		

OFC	:ADT:NRR	:DD:NRR	:Tech Editor			
NAME	:WTRussell	:FMiraglia	:JMain*			
DATE	:11/21/90	:11/28/90	:11/14/90			

OFFICIAL RECORD COPY DOCUMENT NAME: JORDAN MEMO

11/23/90

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Enclosure: see next page
1. CRGR Review Package
2. Draft generic letter on Licensee
   Commercial-Grade Procurement
   and Dedication Programs

CONTACT: Richard P. McIntyre, NRR
492-3215

* see previous concurrence

DISTRIBUTION (w/enclosures)

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January 10, 1991

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

SUBJECT: MINUTES OF CRGR MEETING 197

The Committee to Review Generic Requirements (CRGR) met on Tuesday, December 18, 1990 from 2:00 - 5:00 p.m. A list of attendees at the meeting is enclosed (Enclosure 1). The following items were discussed at the meeting.

1. B. Grimes of NRR presented for CRGR review a proposed generic letter on licensee commercial-grade dedication and procurement programs. Although the package stated that it involved no new positions or backfitting, the CRGR expressed the opinion that the package, as presented, seemed to be a backfit. The staff agreed to provide another package, modified so it would not constitute backfitting in the near future. This matter is discussed in Enclosure 2.

In accordance with the EDO's July 18, 1983 directive concerning "Feedback and Closure of CRGR Review," a written response is required from the cognizant office to report agreement or disagreement with CRGR recommendations in these minutes. The response, which is required within five working days after receipt of these minutes, is to be forwarded to the CRGR Chairman and if there is disagreement with CRGR recommendations, to the EDO for decision making.

Questions concerning these meeting minutes should be referred to Dennis Allison (492-4148).

Original Signed by:
Denwood F. Ross

for Edward L. Jordan, Chairman
Committee to Review Generic
Requirements

Enclosures:
As stated

cc w/enclosures:
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SECY
J. Lieberman
P. Norry
D. Williams
Regional Administrators
CRGR Members

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G. Mizuno
D. Ross
D. Allison
E. Jordan
J. Conran

ENCLOSURE 1

ATTENDENCE LIST

CRGR Meeting No. 197

December 18, 1990

CRGR Members

E. Jordan
G. Arlotto
F. Miraglia
B. Sheron
L. Reyes
L. J. Callan

CRGR Staff

J. Conran
D. Allison
D. Ross

NRC Staff

W. Russell
B. Grimes
W. Brach
G. Cwalina
E. McKenna
E. Baker
A. Herdt
U. Potapovs
R. McIntyre
C. Vandenburgh

Enclosure 2 to the Minutes of CRGR Meeting No. 197

Proposed Generic Letter on Licensee

Commercial-Grade Dedication and Procurement Programs

December 18, 1990

TOPIC

B. Grimes of NRR presented for CRGR review a proposed generic letter on licensee commercial-grade dedication and procurement programs. The staff had recently instituted a pause in inspection in this area in order to allow time for licensees to improve their programs in accordance with an industry initiative. When inspection activities were resumed, they would initially consist of assessments to determine that a substantive improvement effort was underway. The purposes of the proposed generic letter were to: (1) announce (or confirm) the staff's recent pause in inspections; (2) describe the staff's enforcement practices; and, (3) discuss misunderstandings or weaknesses found in NRC inspections. The package stated that it involved no new positions or backfitting.

BACKGROUND

The review package was transmitted by a memorandum for E. Jordan from F. Miraglia dated November 28, 1990. The package included:

(1) CRGR review package (answers to standard questions)
(2) Draft generic letter

A revised draft generic letter was provided for discussion at the meeting. A copy is provided as Attachment 1 to this enclosure.

The CRGR also received comments from the Nuclear Management and Resources Council (NUMARC) which were distributed at the meeting. A copy is provided in Attachment 2 to this enclosure.

CONCLUSIONS/RECOMMENDATIONS

The CRGR expressed the opinion that the package, as presented, seemed to be a backfit and, unless modified, it should be justified as a backfit.

A primary contributor to this opinion was the enclosure to the generic letter which described weaknesses and misunderstanding found in previous inspections. This appeared to be conveying new staff positions. Further, it appeared to go beyond the industry initiative which had been endorsed by the staff, with some conditions, as an acceptable approach. Finally, the package could appear contradictory - implying that licensees should meet all the recommendations of the industry initiative (and the enclosure) but at the same time maintaining that there were no new positions and the staff's only enforcement standard was Appendix B to 10 CFR 50.

The CRGR expressed the opinion that the package could be modified so it would not constitute backfitting. The primary modification would be deleting or substantially modifying the enclosure which discussed weaknesses and misunderstandings found in the previous inspections. In this case, the CRGR would support issuance of the generic letter, subject to CRGR staff check of the revised letter (and possibly circulating the revised letter to the members). The staff agreed to provide a revised package along these lines in the near future.

It was noted that the CRGR wanted to see the procedures for the forthcoming assessments to determine that a substantive improvement effort was underway. The staff agreed to provide the procedures when they were written.

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

mta

DRAFT

TO: ALL HOLDERS OF OPERATING LICENSES AND CONSTRUCTION PERMITS FOR NUCLEAR POWER REACTORS

SUBJECT: LICENSE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS (90-XX)

This generic letter notifies the industry of the staff's intent to pause in conducting certain procurement inspection and enforcement activities and identifies a number of failures in the licensees' commercial-grade dedication programs identified during recent team inspections performed by the U.S. Nuclear Regulatory Commission (NRC). This generic letter provides discussion of the staff's views on key activities which, if they had been included in licensee implementation of these programs, could have avoided such failures.

During the period from 1986 to 1989, the NRC conducted 13 team inspections of the licensees' procurement and commercial-grade dedication programs. During these inspections, the NRC staff identified a common, programmatic deficiency in the licensees' control of the procurement and dedication process of commercial-grade items for safety-related applications. In a number of cases, the staff found that licensees had failed to adequately maintain programs to assure the suitability of commercially procured and dedication equipment for its intended safety-related applications. In addition, the staff identified equipment of indeterminate quality installed in the licensee's facilities.

The NRC staff believes from these inspection findings that, there has been a change in the industry's procurement practices and a decrease in the number of qualified nuclear-grade vendors. Ten years ago, licensees procured major assemblies from approved vendors who maintained quality assurance programs pursuant to Appendix B of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR). Currently, due to the reduction in the number of qualified nuclear-grade vendors, licensees are increasing the numbers of commercial-grade replacement parts that they procure and dedicate for use in safety-related applications. This has resulted in an increased emphasis by the staff that licensees maintain dedication programs that assure the quality of items purchased. Therefore, dedication processes for commercial-grade parts have increased in importance and NRC inspections have determined that a number of licensees have not satisfactorily performed this procurement and dedication process.

The industry has been made fully aware of the NRC's concerns in this program area. In the past, escalated enforcement cases have provided notice to the affected licensees and to the industry of NRC's findings, concerns, and expectations in the implementation of procurement and dedication programs. Further, the NRC staff continues to participate in numerous industry meetings and conferences at which the NRC's positions in this area have been presented.

ATTACHMENT 1 TO
ENCLOSURE 2

The Nuclear Utility Management and Resources Council (NUMARC) Board of Directors recently approved a comprehensive procurement initiative. While monitoring industry implementation of licensee program improvements, the NRC staff is deferring inspections of licensees' procurement and commercial-grade dedication processes for about a year to allow utilities sufficient time to fully understand and implement the guidance being developed by the industry and to evaluate the effectiveness of the programs.

However, the NRC will continue to perform certain types of inspection activities. For example, the staff will conduct selected assessments to determine the progress of the industry in improving the procurement and dedication processes. The staff will continue to perform reactive inspections relating to operational events or to defective equipment and, as required, will continue to initiate resultant enforcement actions which will not be affected by the decision to defer programmatic inspections. In addition, the staff will continue to perform inspections of vendors. The staff expects to resume procurement inspection activities in the late summer of 1991.

The staff will not initiate enforcement action in cases of past programmatic violations that have been adequately corrected. In addition, the staff does not expect licensees to review all past procurements. However, if during current procurement activities, licensees identify shortcomings in the form, fit, or function of specific vendor products, or if failure experience or current information on supplier adequacy indicates that a component may not be suitable for service, corrective actions should include a look-back for all such installed and stored items. In performing these actions for both stored and installed items, licensees should follow the existing requirements for corrective and follow-up actions contained in Criterion XVI of 10 CFR Part 50, Appendix B. A licensee should determine programmatic root causes when actual deficiencies in several different vendor products are identified during current procurement activities and when these deficiencies lead to the replacement of installed or warehouse items as part of corrective action. In such cases, a further sampling of previously procured commercial-grade items may be warranted.

In NRC Generic Letter 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," the staff described its perspective on good practices in procurement and provided the NRC's conditional endorsement of an industry standard (EPRI NP-5652) on methods of commercial-grade procurement and dedication. A number of recent inspection findings indicate that licensees have failed to include certain key activities as appropriate in the implementation of the dedication process. Enclosure 1 includes further discussion of the NRC staff's views on the successful implementation of licensees' programs for commercial-grade dedication. The commercial-grade dedication approaches discussed in Enclosure 1 do not constitute new NRC requirements or positions, but rather are intended to ensure a common understanding of implementation issues in this area.

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Although no response to this letter is required, if you have any questions regarding this matter, please contact the persons listed below.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Characteristics of Effective Commercial-Grade Procurement and Dedication Programs
2. List of Recently Issued generic letters

Technical Contact: Richard P. McIntyre, NRR
(301) 492-3215

Uldis Potapovs, NRR
(301) 492-0959

DRAFT

Enclosure 1

CHARACTERISTICS OF EFFECTIVE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS

Background

Appendix B to 10 CFR Part 50 contains the NRC's regulations for procurement quality assurance (QA) and quality control (QC) for products to be used in safety-related applications. In addition, the NRC has provided further guidance in Regulatory Guides 1.28, 1.33, and 1.123. These requirements and guides assure the suitability of equipment, including commercial-grade items for use in safety-related systems. Criterion III of Appendix B requires licensees to select and review for suitability of application materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems, and components. Criterion IV requires that procurement documents specify the applicable requirements necessary to ensure functional performance. Criterion VII requires licensees to assure that the following are sufficient to identify whether specification requirements for the purchased material and equipment have been met: source evaluation and selection, objective evidence of quality, inspection of the source, and examination of products upon delivery. The process used to satisfy these requirements when upgrading commercial-grade items for safety-related applications is commonly called "dedication." The process of ensuring compliance with 10 CFR Part 50, Appendix B, must include all those activities necessary to establish and confirm the quality and suitability of those items to be installed in safety-related applications. Some of the dedication activities may occur early in the procurement cycle, before the item is accepted from the manufacturer. (10 CFR Part 21 has a more restricted definition of commercial-grade item dedication related to responsibility for evaluation and reporting of defects.) Generic Letter (GL) 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," discussed commercial-grade dedication in terms of engineering involvement in the procurement process, product acceptance, and the dedication process as identified in the EPRI NP-5652 guidelines. This enclosure provides examples of specific failures by licensees to fully implement certain key activities for dedicating and ensuring the suitability of commercial-grade products for safety-related applications. Appropriate implementation of these key activities would have avoided the failures in procurement and commercial-grade dedication observed during past NRC inspections.

Inspection Observations and Findings

From 1986 to 1989, headquarters and regional personnel conducted 13 team inspections of licensees' procurement and dedication programs. These inspections have identified a common, broad programmatic deficiency in licensees' control over the process of procurement and dedication of commercial-grade items. In a number of cases, licensees have not maintained

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programs to ensure the suitability of equipment for use in safety-related applications as required by 10 CFR Part 50, Appendix B, Criterion III. From these 13 inspections, the staff identified 8 findings that were considered to be Severity Level III violations and 3 findings that were Severity Level IV violations. At one plant, the staff did not assign a severity level to individual violations. Instead, the staff considered the entire group to be a Severity Level III problem and used enforcement discretion, as provided under the shutdown policy, based on the licensee's corrective actions (see 10 CFR Part 2, Appendix C, Section V.G.2). Only one of the plants that were inspected did not receive violations in this program area.

In GL 89-02, the NRC has conditionally endorsed the dedication methods described in EPRI NP-5652 guidelines. The staff believes that licensees who implement these dedications methods, in accordance with the NRC's endorsement, can establish a basis for satisfying the existing requirements of Appendix B to 10 CFR Part 50 as these requirements apply to the dedication process for commercial-grade items. An effective commercial-grade dedication program should include provisions to demonstrate that a dedicated item is suitable for safety-related applications. For a licensee to adequately establish suitability, certain key activities must be performed as appropriate as part of the dedication process.

During each of the 13 inspections, the staff identified a common element in each of the inspection findings. This element was the failure of the licensee to assure that a commercially procured and dedicated item was suitable for the intended safety-related application. In its ability to perform its intended safety function, a dedicated commercial-grade item should be equivalent to the same item procured under a 10 CFR Part 50, Appendix B QA program. The following is a list of the 13 licensees inspected and the inspection report numbers. A summary of the general inspection findings and NRC observations on these findings follows the list of licensee inspections.

LICENSEE and PLANT	INSPECTION REPORT NO.
1. Tennessee Valley Authority (Sequoyah)	50-327/86-61 50-328/86-61
2. Southern California Edison (San Onofre)	50-206/87-02 50-361/87-03 50-362/87-04
3. Alabama Power (Farley)	50-348/87-11 50-364/87-11
4. Louisiana Power and Light (Waterford)	50-382/87-19

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LICENSEE and PLANT		INSPECTION REPORT NO.
5.	Sacramento Municipal Utility District (Rancho Seco)	50-312/88-02
6.	Maine Yankee Atomic Power (Maine Yankee)	50-309/88-200
7.	Northern States Power (Prairie Island)	50-282/88-201 50-306/88-201
8.	Portland General Electric (Trojan)	50-344/88-39 50-344/88-46
9.	Connecticut Yankee Atomic Power (Haddam Neck)	50-213/89-200
10.	Washington Public Power Supply System (WNP-2)	50-397/89-21 50-397/89-28
11.	Florida Power (Crystal River)	50-302/89-200
12.	Gulf States Utilities (River Bend)	50-458/89-200
13.	Commonwealth Edison (Zion)	50-295/89-200 50-304/89-200

1. Inspection Findings

   a. Failure to identify the methods and acceptance criteria for verifying the critical characteristics, such as during receipt inspection, dedication process, or post-installation testing.

   b. Failure to establish verifiable, documented traceability of complex commercial-grade items to their original equipment manufacturers in those cases where the dedication program cannot verify the critical characteristics.

   c. Failure to recognize that some commercial-grade items cannot be fully dedicated once received on site. Certain items are manufactured using special processes, such as welding and heat treating. Dedication testing of these items as finished products would destroy them. For these items, licensees may need to conduct vendor surveillances or to witness certain activities during the manufacturing process.

Discussion

The NRC staff has met on several occasions with NUMARC and licensee representatives to discuss "critical characteristics" as used in the context of commercial-grade procurement and dedication. The term "critical

characteristics" is not contained in Appendix B and has no special regulatory significance beyond its use and definition in various industry guides and standards. The NRC has not taken the position that all design requirements must be considered to be critical characteristics as defined and used in EPRI NP-5652. Rather, as stated in Appendix B, Criterion III, licensees must assure the suitability of all parts, materials, and services for their intended safety-related applications (i.e., there needs to be assurance that the item will perform its intended safety function when required). The licensee is responsible for identifying the important design, material, and performance characteristics for each part, material, and service intended for safety-related applications, establishing acceptance criteria, and providing reasonable assurance of the conformance of items to these criteria. There is no minimum or maximum number of critical characteristics that need to be verified. Further, the critical characteristics for an item may vary from application to application depending on the design and performance requirements unique to each application.

A licensee may take different approaches for the verification of the critical characteristics, depending on the complexity of the item. In many cases, the licensee can verify the critical characteristics of a simple item during the receipt inspection. However, for a complex item with internal parts which receive special processing during manufacturing, the licensee would probably need to audit or survey the vendor to verify the critical characteristics necessary for the item to perform its safety function. When the dedication program cannot verify the critical characteristics related to special processes and tests, the licensee should establish documented, verifiable traceability to the original equipment manufacturer. For simple items with critical characteristics that can be verified for the most severe or limiting plant application, the licensee might prefer a broad dedication program to identify and verify the item's critical characteristics to qualify that item for all possible plant applications. For complex items that would be purchased for specific plant applications, the licensee should address the acceptance criteria for each item individually. Engineering involvement is essential in either method because the technical evaluation will identify the critical characteristics, acceptance criteria, and the methods to be used for verification.

2. Inspection Findings

a. Failure to demonstrate that a like-for-like replacement item is identical in form, fit, and function to the item it is replacing. Part number verification is not sufficient because of the probability of undocumented changes in the design, material, or fabrication of commercial-grade items using the same part number.

b. Failure to evaluate changes in the design, material, or manufacturing process for the effect of these changes on safety function performance (particularly under design basis event conditions) of replacement items that are similar as opposed to identical to the items being replaced.

c. Failure to ensure that items will function under all design requirements. On some occasions, licensees only ensured that the commercial-grade item would function under normal operation conditions.

d. Failure to verify the validity of certificates of conformance received from vendors not on the licensee's list of approved vendors/suppliers. An unverified certificate of conformance from a commercial-grade vendor is not sufficient.

Discussion

A like-for-like replacement is defined as the replacement of an item with an item that is identical. A like-for-like replacement does not change the engineering analysis or as-built configuration of the component or system in which it is installed, and the replacement item meets the same design specifications, technical and quality requirements, and functional characteristics as the item it replaces. If differences from the original item are identified in the replacement item, then the item is not identical, but similar to the item being replaced, and an evaluation must be performed to determine if any changes in design, material, or the manufacturing process could impact the functional characteristics and ultimately the component's ability to perform its required safety function.

If the licensee can demonstrate that the replacement item is identical, then the licensee need not identify the safety function or review and verify the design requirements and critical characteristics. For example, the replacement item would be identical if it was purchased at the same time from the same vendor as the item it is replacing, or if the user can verify that there have been no changes in the design, materials, or manufacturing process since procurement of the item being replaced.

Engineering involvement is essential in the above activities. The extent of this involvement is dependent on the nature, complexity, and use of the items to be dedicated. Engineering personnel should participate in the procurement process, and product acceptance, to develop purchase specifications, determine specific testing requirements applicable to the products, and evaluate the test results. When engineering personnel specify design requirements for inclusion on the purchase documents for replacement components, they need not reconstruct and reverify the design adequacy, but only ensure that these design requirements (which may reference the original design basis) are properly translated into the purchase order.

Reliance on part number verification and certification documentation is insufficient to ensure the quality of commercially procured products. To conduct effective product acceptance programs, licensees should ensure that these programs include receipt and source inspection, appropriate testing criteria, effective vendor audits (including witness/hold points), special tests and inspections, and post-installation tests. The licensees should establish procedures to implement their programs and should ensure that the implementing personnel have adequate qualifications and training.

Rec'd by fax in CRGR offices Mon ~ 5:00pm

NUMARC

NUCLEAR MANAGEMENT AND RESOURCES COUNCIL

1776 Eye Street, N.W. • Suite 300 • Washington, DC 20006-2496
(202) 872-1280

December 17, 1990

Mr. Edward L. Jordan, Chairman
Committee to Review Generic Requirements
U. S. Nuclear Regulatory Commission
Mail Stop 3701
Washington, D. C. 20555

Dear Mr. Jordan:

We understand that CRGR is currently reviewing a draft Generic Letter (GL) addressing dedication of commercial grade items for safety related application and potentially other issues related to industry procurement improvement programs. An earlier version of this draft GL was provided to NUMARC for industry review on April 27, 1990. Our letter of May 16, 1990 to Mr. Brian Grimes provided comments on that earlier version. We understand that the draft GL has been significantly revised since that time, but NRC has not provided later versions for our review, so we are unaware of the content of the draft GL currently undergoing CRGR review.

We have met with the NRC staff on numerous occasions to discuss the issues addressed by the earlier version of the draft GL. Based on these discussions, it is unclear whether the positions delineated in our May 16 letter have been taken into account in revising the draft Generic Letter. We would like to reiterate that the positions expressed in that letter remain valid and should be give careful consideration.

Moreover, in discussions with the staff since our comments on the earlier version of the draft GL, and in review of SECY 90-304, which has been issued in the interim, we have identified some concerns with an additional NRC staff position that may be addressed in the current draft GL. This position involves the relationship of 10 CFR 50 Appendix B requirements to the industry initiatives developed to bring about industry procurement program improvements. The improvements developed by the industry were not considered as additional mechanisms to address the requirements of Appendix B. Each utility already has an NRC approved Appendix B program. Rather, the improvements were developed to be utilized by utilities to address fundamental changes in the marketplace. These changes include: 1) A primary need for replacement piece parts rather than complete equipment; 2) Diminishing numbers of vendors with Appendix B programs and resultant need for use of commercial grade items; and, 3) Increased obsolescence of installed items. NRC officials have stated that no widespread or significant safety issues have been found due to existing procurement practices. We believe prescriptive regulatory approaches to the industry-initiated improvements are therefore unwarranted. We are hopeful that the initiatives will not be viewed, either by industry or NRC, as an "extension" of Appendix B type requirements into new areas and affecting additional vendors. This will only exacerbate the situation that has led to diminished numbers of quality vendors and to related problems.

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ATTACHMENT 2 TO
ENCLOSURE 2

We appreciate the opportunity to provide comments to CRGR on this important issue. We would be happy to provide any additional input that NRC may find useful in addressing our mutual need to assure the continuation of high quality in the items procured by the industry. Please contact me if you have any further questions.

Sincerely,

William H. Rasin
Director, Technical Division

REB/

cc: Brian K. Grimes, NRR

January 10, 1991

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

SUBJECT: MINUTES OF CRGR MEETING 197

The Committee to Review Generic Requirements (CRGR) met on Tuesday, December 18, 1990 from 2:00 - 5:00 p.m. A list of attendees at the meeting is enclosed (Enclosure 1). The following items were discussed at the meeting.

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Questions concerning these meeting minutes should be referred to Dennis Allison (492-4148).

Original Signed by:
Denwood F. Ross

for Edward L. Jordan, Chairman
Committee to Review Generic Requirements

Enclosures:
As stated

cc w/enclosures:
Commission (5)
SECY
J. Lieberman
P. Norry
D. Williams
Regional Administrators
CRGR Members

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G. Mizuno
D. Ross
D. Allison
E. Jordan
J. Conran

ENCLOSURE 1

ATTENDENCE LIST

CRGR Meeting No. 197

December 18, 1990

CRGR Members

E. Jordan
G. Arlotto
F. Miraglia
B. Sheron
L. Reyes
L. J. Callan

CRGR Staff

J. Conran
D. Allison
D. Ross

NRC Staff

W. Russell
B. Grimes
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E. McKenna
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A. Herdt
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R. McIntyre
C. Vandenburgh

Enclosure 2 to the Minutes of CRGR Meeting No. 197
Proposed Generic Letter on Licensee
Commercial-Grade Dedication and Procurement Programs

December 18, 1990

TOPIC

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(2) Draft generic letter

A revised draft generic letter was provided for discussion at the meeting. A copy is provided as Attachment 1 to this enclosure.

The CRGR also received comments from the Nuclear Management and Resources Council (NUMARC) which were distributed at the meeting. A copy is provided in Attachment 2 to this enclosure.

CONCLUSIONS/RECOMMENDATIONS

The CRGR expressed the opinion that the package, as presented, seemed to be a backfit and, unless modified, it should be justified as a backfit.

A primary contributor to this opinion was the enclosure to the generic letter which described weaknesses and misunderstanding found in previous inspections. This appeared to be conveying new staff positions. Further, it appeared to go beyond the industry initiative which had been endorsed by the staff, with some conditions, as an acceptable approach. Finally, the package could appear contradictory - implying that licensees should meet all the recommendations of the industry initiative (and the enclosure) but at the same time maintaining that there were no new positions and the staff's only enforcement standard was Appendix B to 10 CFR 50.

The CRGR expressed the opinion that the package could be modified so it would not constitute backfitting. The primary modification would be deleting or substantially modifying the enclosure which discussed weaknesses and misunderstandings found in the previous inspections. In this case, the CRGR would support issuance of the generic letter, subject to CRGR staff check of the revised letter (and possibly circulating the revised letter to the members). The staff agreed to provide a revised package along these lines in the near future.

It was noted that the CRGR wanted to see the procedures for the forthcoming assessments to determine that a substantive improvement effort was underway. The staff agreed to provide the procedures when they were written.

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

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TO: ALL HOLDERS OF OPERATING LICENSES AND CONSTRUCTION PERMITS FOR NUCLEAR POWER REACTORS

SUBJECT: LICENSE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS (90-XX)

This generic letter notifies the industry of the staff's intent to pause in conducting certain procurement inspection and enforcement activities and identifies a number of failures in the licensees' commercial-grade dedication programs identified during recent team inspections performed by the U.S. Nuclear Regulatory Commission (NRC). This generic letter provides discussion of the staff's views on key activities which, if they had been included in licensee implementation of these programs, could have avoided such failures.

During the period from 1986 to 1989, the NRC conducted 13 team inspections of the licensees' procurement and commercial-grade dedication programs. During these inspections, the NRC staff identified a common, programmatic deficiency in the licensees' control of the procurement and dedication process of commercial-grade items for safety-related applications. In a number of cases, the staff found that licensees had failed to adequately maintain programs to assure the suitability of commercially procured and dedication equipment for its intended safety-related applications. In addition, the staff identified equipment of indeterminate quality installed in the licensee's facilities.

The NRC staff believes from these inspection findings that, there has been a change in the industry's procurement practices and a decrease in the number of qualified nuclear-grade vendors. Ten years ago, licensees procured major assemblies from approved vendors who maintained quality assurance programs pursuant to Appendix B of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR). Currently, due to the reduction in the number of qualified nuclear-grade vendors, licensees are increasing the numbers of commercial-grade replacement parts that they procure and dedicate for use in safety-related applications. This has resulted in an increased emphasis by the staff that licensees maintain dedication programs that assure the quality of items purchased. Therefore, dedication processes for commercial-grade parts have increased in importance and NRC inspections have determined that a number of licensees have not satisfactorily performed this procurement and dedication process.

The industry has been made fully aware of the NRC's concerns in this program area. In the past, escalated enforcement cases have provided notice to the affected licensees and to the industry of NRC's findings, concerns, and expectations in the implementation of procurement and dedication programs. Further, the NRC staff continues to participate in numerous industry meetings and conferences at which the NRC's positions in this area have been presented.

ATTACHMENT 1 TO
ENCLOSURE 2

The Nuclear Utility Management and Resources Council (NUMARC) Board of Directors recently approved a comprehensive procurement initiative. While monitoring industry implementation of licensee program improvements, the NRC staff is deferring inspections of licensees' procurement and commercial-grade dedication processes for about a year to allow utilities sufficient time to fully understand and implement the guidance being developed by the industry and to evaluate the effectiveness of the programs.

However, the NRC will continue to perform certain types of inspection activities. For example, the staff will conduct selected assessments to determine the progress of the industry in improving the procurement and dedication processes. The staff will continue to perform reactive inspections relating to operational events or to defective equipment and, as required, will continue to initiate resultant enforcement actions which will not be affected by the decision to defer programmatic inspections. In addition, the staff will continue to perform inspections of vendors. The staff expects to resume procurement inspection activities in the late summer of 1991.

The staff will not initiate enforcement action in cases of past programmatic violations that have been adequately corrected. In addition, the staff does not expect licensees to review all past procurements. However, if during current procurement activities, licensees identify shortcomings in the form, fit, or function of specific vendor products, or if failure experience or current information on supplier adequacy indicates that a component may not be suitable for service, corrective actions should include a look-back for all such installed and stored items. In performing these actions for both stored and installed items, licensees should follow the existing requirements for corrective and follow-up actions contained in Criterion XVI of 10 CFR Part 50, Appendix B. A licensee should determine programmatic root causes when actual deficiencies in several different vendor products are identified during current procurement activities and when these deficiencies lead to the replacement of installed or warehouse items as part of corrective action. In such cases, a further sampling of previously procured commercial-grade items may be warranted.

In NRC Generic Letter 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," the staff described its perspective on good practices in procurement and provided the NRC's conditional endorsement of an industry standard (EPRI NP-5652) on methods of commercial-grade procurement and dedication. A number of recent inspection findings indicate that licensees have failed to include certain key activities as appropriate in the implementation of the dedication process. Enclosure 1 includes further discussion of the NRC staff's views on the successful implementation of licensees' programs for commercial-grade dedication. The commercial-grade dedication approaches discussed in Enclosure 1 do not constitute new NRC requirements or positions, but rather are intended to ensure a common understanding of implementation issues in this area.

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Although no response to this letter is required, if you have any questions regarding this matter, please contact the persons listed below.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Characteristics of Effective Commercial-Grade Procurement and Dedication Programs
2. List of Recently Issued generic letters

Technical Contact: Richard P. McIntyre, NRR
(301) 492-3215

Uldis Potapovs, NRR
(301) 492-0959

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Enclosure 1

CHARACTERISTICS OF EFFECTIVE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS

Background

Appendix B to 10 CFR Part 50 contains the NRC's regulations for procurement quality assurance (QA) and quality control (QC) for products to be used in safety-related applications. In addition, the NRC has provided further guidance in Regulatory Guides 1.28, 1.33, and 1.123. These requirements and guides assure the suitability of equipment, including commercial-grade items for use in safety-related systems. Criterion III of Appendix B requires licensees to select and review for suitability of application materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems, and components. Criterion IV requires that procurement documents specify the applicable requirements necessary to ensure functional performance. Criterion VII requires licensees to assure that the following are sufficient to identify whether specification requirements for the purchased material and equipment have been met: source evaluation and selection, objective evidence of quality, inspection of the source, and examination of products upon delivery. The process used to satisfy these requirements when upgrading commercial-grade items for safety-related applications is commonly called "dedication." The process of ensuring compliance with 10 CFR Part 50, Appendix B, must include all those activities necessary to establish and confirm the quality and suitability of those items to be installed in safety-related applications. Some of the dedication activities may occur early in the procurement cycle, before the item is accepted from the manufacturer. (10 CFR Part 21 has a more restricted definition of commercial-grade item dedication related to responsibility for evaluation and reporting of defects.) Generic Letter (GL) 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," discussed commercial-grade dedication in terms of engineering involvement in the procurement process, product acceptance, and the dedication process as identified in the EPRI NP-5652 guidelines. This enclosure provides examples of specific failures by licensees to fully implement certain key activities for dedicating and ensuring the suitability of commercial-grade products for safety-related applications. Appropriate implementation of these key activities would have avoided the failures in procurement and commercial-grade dedication observed during past NRC inspections.

Inspection Observations and Findings

From 1986 to 1989, headquarters and regional personnel conducted 13 team inspections of licensees' procurement and dedication programs. These inspections have identified a common, broad programmatic deficiency in licensees' control over the process of procurement and dedication of commercial-grade items. In a number of cases, licensees have not maintained

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programs to ensure the suitability of equipment for use in safety-related applications as required by 10 CFR Part 50, Appendix B, Criterion III. From these 13 inspections, the staff identified 8 findings that were considered to be Severity Level III violations and 3 findings that were Severity Level IV violations. At one plant, the staff did not assign a severity level to individual violations. Instead, the staff considered the entire group to be a Severity Level III problem and used enforcement discretion, as provided under the shutdown policy, based on the licensee's corrective actions (see 10 CFR Part 2, Appendix C, Section V.G.2). Only one of the plants that were inspected did not receive violations in this program area.

In GL 89-02, the NRC has conditionally endorsed the dedication methods described in EPRI NP-5652 guidelines. The staff believes that licensees who implement these dedications methods, in accordance with the NRC's endorsement, can establish a basis for satisfying the existing requirements of Appendix B to 10 CFR Part 50 as these requirements apply to the dedication process for commercial-grade items. An effective commercial-grade dedication program should include provisions to demonstrate that a dedicated item is suitable for safety-related applications. For a licensee to adequately establish suitability, certain key activities must be performed as appropriate as part of the dedication process.

During each of the 13 inspections, the staff identified a common element in each of the inspection findings. This element was the failure of the licensee to assure that a commercially procured and dedicated item was suitable for the intended safety-related application. In its ability to perform its intended safety function, a dedicated commercial-grade item should be equivalent to the same item procured under a 10 CFR Part 50, Appendix B QA program. The following is a list of the 13 licensees inspected and the inspection report numbers. A summary of the general inspection findings and NRC observations on these findings follows the list of licensee inspections.

LICENSEE and PLANT	INSPECTION REPORT NO.
1. Tennessee Valley Authority (Sequoyah)	50-327/86-61 50-328/86-61
2. Southern California Edison (San Onofre)	50-206/87-02 50-361/87-03 50-362/87-04
3. Alabama Power (Farley)	50-348/87-11 50-364/87-11
4. Louisiana Power and Light (Waterford)	50-382/87-19

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LICENSEE and PLANT		INSPECTION REPORT NO.
5.	Sacramento Municipal Utility District (Rancho Seco)	50-312/88-02
6.	Maine Yankee Atomic Power (Maine Yankee)	50-309/88-200
7.	Northern States Power (Prairie Island)	50-282/88-201 50-306/88-201
8.	Portland General Electric (Trojan)	50-344/88-39 50-344/88-46
9.	Connecticut Yankee Atomic Power (Haddam Neck)	50-213/89-200
10.	Washington Public Power Supply System (WNP-2)	50-397/89-21 50-397/89-28
11.	Florida Power (Crystal River)	50-302/89-200
12.	Gulf States Utilities (River Bend)	50-458/89-200
13.	Commonwealth Edison (Zion)	50-295/89-200 50-304/89-200

1. Inspection Findings

a. Failure to identify the methods and acceptance criteria for verifying the critical characteristics, such as during receipt inspection, dedication process, or post-installation testing.

b. Failure to establish verifiable, documented traceability of complex commercial-grade items to their original equipment manufacturers in those cases where the dedication program cannot verify the critical characteristics.

c. Failure to recognize that some commercial-grade items cannot be fully dedicated once received on site. Certain items are manufactured using special processes, such as welding and heat treating. Dedication testing of these items as finished products would destroy them. For these items, licensees may need to conduct vendor surveillances or to witness certain activities during the manufacturing process.

Discussion

The NRC staff has met on several occasions with NUMARC and licensee representatives to discuss "critical characteristics" as used in the context of commercial-grade procurement and dedication. The term "critical

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characteristics" is not contained in Appendix B and has no special regulatory significance beyond its use and definition in various industry guides and standards. The NRC has not taken the position that all design requirements must be considered to be critical characteristics as defined and used in EPRI NP-5652. Rather, as stated in Appendix B, Criterion III, licensees must assure the suitability of all parts, materials, and services for their intended safety-related applications (i.e., there needs to be assurance that the item will perform its intended safety function when required). The licensee is responsible for identifying the important design, material, and performance characteristics for each part, material, and service intended for safety-related applications, establishing acceptance criteria, and providing reasonable assurance of the conformance of items to these criteria. There is no minimum or maximum number of critical characteristics that need to be verified. Further, the critical characteristics for an item may vary from application to application depending on the design and performance requirements unique to each application.

A licensee may take different approaches for the verification of the critical characteristics, depending on the complexity of the item. In many cases, the licensee can verify the critical characteristics of a simple item during the receipt inspection. However, for a complex item with internal parts which receive special processing during manufacturing, the licensee would probably need to audit or survey the vendor to verify the critical characteristics necessary for the item to perform its safety function. When the dedication program cannot verify the critical characteristics related to special processes and tests, the licensee should establish documented, verifiable traceability to the original equipment manufacturer. For simple items with critical characteristics that can be verified for the most severe or limiting plant application, the licensee might prefer a broad dedication program to identify and verify the item's critical characteristics to qualify that item for all possible plant applications. For complex items that would be purchased for specific plant applications, the licensee should address the acceptance criteria for each item individually. Engineering involvement is essential in either method because the technical evaluation will identify the critical characteristics, acceptance criteria, and the methods to be used for verification.

2. Inspection Findings

a. Failure to demonstrate that a like-for-like replacement item is identical in form, fit, and function to the item it is replacing. Part number verification is not sufficient because of the probability of undocumented changes in the design, material, or fabrication of commercial-grade items using the same part number.

b. Failure to evaluate changes in the design, material, or manufacturing process for the effect of these changes on safety function performance (particularly under design basis event conditions) of replacement items that are similar as opposed to identical to the items being replaced.

c. Failure to ensure that items will function under all design requirements. On some occasions, licensees only ensured that the commercial-grade item would function under normal operation conditions.

d. Failure to verify the validity of certificates of conformance received from vendors not on the licensee's list of approved vendors/suppliers. An unverified certificate of conformance from a commercial-grade vendor is not sufficient.

Discussion

A like-for-like replacement is defined as the replacement of an item with an item that is identical. A like-for-like replacement does not change the engineering analysis or as-built configuration of the component or system in which it is installed, and the replacement item meets the same design specifications, technical and quality requirements, and functional characteristics as the item it replaces. If differences from the original item are identified in the replacement item, then the item is not identical, but similar to the item being replaced, and an evaluation must be performed to determine if any changes in design, material, or the manufacturing process could impact the functional characteristics and ultimately the component's ability to perform its required safety function.

If the licensee can demonstrate that the replacement item is identical, then the licensee need not identify the safety function or review and verify the design requirements and critical characteristics. For example, the replacement item would be identical if it was purchased at the same time from the same vendor as the item it is replacing, or if the user can verify that there have been no changes in the design, materials, or manufacturing process since procurement of the item being replaced.

Engineering involvement is essential in the above activities. The extent of this involvement is dependent on the nature, complexity, and use of the items to be dedicated. Engineering personnel should participate in the procurement process, and product acceptance, to develop purchase specifications, determine specific testing requirements applicable to the products, and evaluate the test results. When engineering personnel specify design requirements for inclusion on the purchase documents for replacement components, they need not reconstruct and reverify the design adequacy, but only ensure that these design requirements (which may reference the original design basis) are properly translated into the purchase order.

Reliance on part number verification and certification documentation is insufficient to ensure the quality of commercially procured products. To conduct effective product acceptance programs, licensees should ensure that these programs include receipt and source inspection, appropriate testing criteria, effective vendor audits (including witness/hold points), special tests and inspections, and post-installation tests. The licensees should establish procedures to implement their programs and should ensure that the implementing personnel have adequate qualifications and training.

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NUMARC

NUCLEAR MANAGEMENT AND RESOURCES COUNCIL

1776 Eye Street, N.W. • Suite 300 • Washington, DC 20006-2496
(202) 872-1280

Rec'd by fax in CRGR office Mon ~ 5:00pm

December 17, 1990

Mr. Edward L. Jordan, Chairman
Committee to Review Generic Requirements
U. S. Nuclear Regulatory Commission
Mail Stop 3701
Washington, D. C. 20555

Dear Mr. Jordan:

We understand that CRGR is currently reviewing a draft Generic Letter (GL) addressing dedication of commercial grade items for safety related application and potentially other issues related to industry procurement improvement programs. An earlier version of this draft GL was provided to NUMARC for industry review on April 27, 1990. Our letter of May 16, 1990 to Mr. Brian Grimes provided comments on that earlier version. We understand that the draft GL has been significantly revised since that time, but NRC has not provided later versions for our review, so we are unaware of the content of the draft GL currently undergoing CRGR review.

We have met with the NRC staff on numerous occasions to discuss the issues addressed by the earlier version of the draft GL. Based on these discussions, it is unclear whether the positions delineated in our May 16 letter have been taken into account in revising the draft Generic Letter. We would like to reiterate that the positions expressed in that letter remain valid and should be give careful consideration.

Moreover, in discussions with the staff since our comments on the earlier version of the draft GL, and in review of SECY 90-304, which has been issued in the interim, we have identified some concerns with an additional NRC staff position that may be addressed in the current draft GL. This position involves the relationship of 10 CFR 50 Appendix B requirements to the industry initiatives developed to bring about industry procurement program improvements. The improvements developed by the industry were not considered as additional mechanisms to address the requirements of Appendix B. Each utility already has an NRC approved Appendix B program. Rather, the improvements were developed to be utilized by utilities to address fundamental changes in the marketplace. These changes include: 1) A primary need for replacement piece parts rather than complete equipment; 2) Diminishing numbers of vendors with Appendix B programs and resultant need for use of commercial grade items; and, 3) Increased obsolescence of installed items. NRC officials have stated that no widespread or significant safety issues have been found due to existing procurement practices. We believe prescriptive regulatory approaches to the industry-initiated improvements are therefore unwarranted. We are hopeful that the initiatives will not be viewed, either by industry or NRC, as an "extension" of Appendix B type requirements into new areas and affecting additional vendors. This will only exacerbate the situation that has led to diminished numbers of quality vendors and to related problems.

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ATTACHMENT 2 TO
ENCLOSURE 2

We appreciate the opportunity to provide comments to CRGR on this important issue. We would be happy to provide any additional input that NRC may find useful in addressing our mutual need to assure the continuation of high quality in the items procured by the industry. Please contact me if you have any further questions.

Sincerely,

William H. Rasin

William H. Rasin
Director, Technical Division

REB/

cc: Brian K. Grimes, NRR

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

B/file

NOV 28 1990

MEMORANDUM FOR: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

FROM: Frank J. Miraglia, Deputy Director
Office of Nuclear Reactor Regulation

SUBJECT: PROPOSED GENERIC LETTER ON LICENSEE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS

The Office of Nuclear Reactor Regulation requests that the Committee to Review Generic Requirements (CRGR) consider the enclosed proposed generic letter. The staff is proposing the enclosed generic letter to notify the industry of the staff's intent to pause in conducting programmatic procurement inspection and enforcement activities and to identify a number of failures in the licensees' commercial-grade dedication programs identified during past NRC inspections. This generic letter also provides information from the NRC's inspections of the licensees' commercial-grade procurement and dedication programs which, if included in licensees' implementation of these programs, could have avoided violations of regulatory requirements.

The commercial-grade dedication inspection findings discussed in Enclosure 1 to the generic letter are based upon 10 CFR Part 50, Appendix B requirements and do not involve changes in the staff's positions. Further, the proposed generic letter does not require any specific licensee action or response to the NRC based on the issuance of this generic letter. Because no new regulations or regulatory practices are involved, the relation to the Commission's safety goals have not been explicitly addressed. However, this action appears to relate to how well a plant is operated. The matters addressed in this generic letter contribute to reducing or avoiding a substantial increase in uncertainty in the assumptions on which safety goal calculations are based.

Enclosure 2 to this memorandum is the proposed generic letter and Enclosure 1 contains the CRGR review package. Brian K. Grimes, Director, Division of Reactor Inspection and Safeguards, is the sponsoring division director. OGC concurrence is currently being sought.

Frank J. Miraglia

Frank J. Miraglia, Deputy Director
Office of Nuclear Reactor Regulation

Enclosure:
1. CRGR Review Package
2. Draft Generic Letter on Licensee Commercial-Grade Procurement and Dedication Programs

CONTACT: Richard P. McIntyre, NRR
492-3215

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ENCLOSURE 1

CRGR REVIEW PACKAGE

RESPONSE TO REQUIREMENTS FOR CONTENT OF PACKAGE SUBMITTED FOR CRGR REVIEW

(i) The proposed generic requirement as it is proposed to be sent out to all holders of operating licenses and construction permits for nuclear power plants.

The staff position is:

The proposed position is stated in the proposed generic letter. In summary, all holders of operating licenses and construction permits for nuclear power reactors would be notified of the staff's intent to pause in conducting programmatic procurement inspection and enforcement activities. However, the NRC will conduct selected assessments to determine the progress of the industry in improving procurement and dedication programs. (Utilities are now implementating the Nuclear Management Resources Council (NUMARC) Initiative on the Dedication of Commercial-Grade Items and the Comprehensive Procurement Initiative). This generic letter identifies a number of failures in the licensees' commercial-grade dedication programs that were identified during recent NRC inspections. In addition, this generic letter provides the staff's views on key activities, which, if included in licensee implementation of these programs, could have avoided such failures.

(ii) Draft staff papers or other underlying staff documents supporting the requirements or staff (regulatory) positions. (A copy of all materials referenced in the document shall be made available upon request to the CRGR staff. Any committee member may request that the CRGR staff obtain a copy of any referenced material for his or her use.)

The following documents support the staff's position:

(a) Proposed NRC Generic Letter 90-XX: "Licensee Commerical-Grade Procurement and Dedication Programs" (See generic letter in Enclosure 2).

(b) Enclosure 1 of the proposed generic letter lists 13 NRC Inspection Reports regarding licensees' procurement and dedication programs.

(c) NRC Generic Letter 89-02: "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products." Enclosure 1 to Generic Letter 89-02 lists NRC bulletins and information notices regarding nonconforming materials and equipment.

(d) SECY-90-057, Advance Notice of Proposed Rulemaking, "Acceptance of Products Purchased for Use in Nuclear Power Plant Structures, Systems, and Components."

(e) SECY-90-304, "NUMARC Initiatives on Procurement."

(f) SECY-90-261, "Inspection and Enforcement Initiatives for Commercial-Grade Procurement and Dedication Programs."

(iii) Each proposed requirement or staff (regulatory) position shall contain the sponsoring office's position as to whether the proposal would increase requirements or staff (regulatory) positions, implement existing requirements or staff (regulatory) positions, or would relax or reduce existing requirements or staff (regulatory) positions.

The commercial-grade dedication approaches discussed in Enclosure 1 of the proposed generic letter do not constitute new NRC requirements or positions, but provide specific clarifications to implementation guidance to meet 10 CFR Part 50, Appendix B. However, if current or improved procurement activities identify shortcomings in the form, fit, or function of specific vendor products or if failure experience or current information on supplier adequacy indicates that a component may not be suitable for service, corrective actions should include a look-back for all such installed and stored items. The licensees' actions in this regard for both warehouse and installed items should follow the existing requirements for corrective action and follow-up contained in Criterion XVI of 10 CFR Part 50, Appendix B.

(iv) The proposed method of implementation along with the concurrence (and any comments) of OGC on the method proposed.

The staff proposes to promulgate the clarification by means of a generic letter. This method has been effective in the past. The Office of the General Counsel (OGC) has provided comments and has concurred in the proposed generic letter.

(v) Regulatory analyses generally conforming to the directives and guidance of NUREG/BR-0058 and NUREG/CR-3568. (Make sufficient to address the Paperwork Reduction Act, the Regulatory Flexibility Act, and Executive Order 12291).

(a) This request for information was approved by the Office of Management and Budget under blanket clearance number 3150-0011 as meeting the requirements of the Paper Reduction Act and Executive Order 12291.

(b) Because this request is not a rulemaking action, the Regulatory Flexibility Act does not apply.

(vi) Identification of the category of reactor plants to which the generic requirement or staff position is to apply (that is, whether it is to apply to new plants only, new OLs only, OLs after a certain date, OLs before a certain date, all OLs, all plants under construction, all plants, all water reactors, all PWRs only, some vendor types, some vintage types such as BWR 6 and 4, jet pump and nonjet pump plants, etc.)

As described in Item (i) above, the proposed requirements apply to all holders of operating licenses and construction permits for nuclear power reactors.

(vii) For each such category of reactor plants, an evaluation which demonstrates how the action should be prioritized and scheduled in light of other ongoing regulatory activities. The evaluation shall document for consideration information available concerning any of the proposed backfit factors as may be appropriate and any other information relevant and material to the proposed action:

Respo to this item is not required pursuant to Revision 4 of the CRGR Charter, Section III.D., because the proposed generic letter announces an NRC inspection pause and conforming to the staff views on key dedication activities would bring licensees into compliance with existing regulatory requirements. This action should not affect the industry's schedule for improvements because the initiative on commerical-grade dedication was implemented in early 1990 and the comprehensive procuremen nitiative is already underway.

(viii) For each evaluation conducted pursuant to 10 CFR 50.109, the proposing office director's determination, together with the rationale for the determination based on the considerations of paragraphs (i) through (vii) above, that:

(a) There is a substantial increase in the overall protection of public health and safety or the common defense and security to be derived from the proposal; and

This item is not applicable since no changes in staff positions are involved. However, the following discussion provides the safety significance of this action:

The NRC has identified numerous instances in which the nuclear industry received, accepted, and installed products that were not of the quality identified by the manufacturer or supplier. The NRC has also identified examples of significant deficiencies in the procurement and dedication of commercial-grade items, with errors traceable to both suppliers and purchasers who dedicate the items for safety-related applications.

The inadequate dedication of commercial-grade items by suppliers and purchasers (including licensees), increases the probability that hardware installed in safety-related applications may not perform as desired. Therefore, the guidance in the proposed generic letter provides for overall protection of public health and safety.

The NUMARC Initiative on the Dedication of Commercial-Grade Items requested that utilities review and, if necessary, develop or upgrade current programs to meet the intent of Electric Power Research Institute (EPRI) NP-5652. Generic Letter 89-02 conditionally endorses EPRI NP-5652 as a guideline for commercial-grade dedication. The EPRI guideline presents several approaches to implement existing requirements as they apply to commercial-grade items.

(b) The direct and indirect costs of implementation, for the facilities affected, are justified in view of this increased protection.

(1) Direct and indirect costs associated with the required actions by the generic letter result primarily from the evaluation by licensees of their existing procurement programs, and, for deficient programs, the necessary corrective actions. The licensees are performing this review as a result of the NUMARC initiative and should not require substantial additional resources in order to consider the staff views expressed in the generic letter.

The amount of effort needed to correct deficient programs will be a function of the current adequacy of licensee's programs and may range from no changes to changes that require several FTEs each year. The staff believes that the costs of implementation are justified in view of the need to ensure the suitability of materials and equipment procured for use in nuclear safety-related applications.

(2) Occupational radiation exposure should not increase because of the actions requested by this generic letter.

(3) NRC resources will be required to conduct selected assessments to determine the progress of the industry in implementation of the initiative on the dedication of commercial-grade items.

(ix) For each evaluation conducted for proposed relaxations or decreases in current requirements of staff positions, the action is justified because of the proposing office director's determination, together with the rationale for the determination based on the considerations of the above, that:

(a) the public health and safety and the common defense and security would be adequately protected if the proposed reduction in requirements or (regulatory) positions were implemented; and

(b) the cost savings attributed to the action would be substantial enough to justify taking the action.

This item is not applicable to the proposed generic letter because the staff is not proposing a relaxation or decrease in current requirements.

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

ENCLOSURE 2

TO: ALL HOLDERS OF OPERATING LICENSES AND CONSTRUCTION PERMITS FOR NUCLEAR POWER REACTORS

SUBJECT: LICENSEE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS (GL 90-XX)

This generic letter notifies the industry of the staff's intent to pause in conducting certain procurement inspection and enforcement activities and to identify a number of failures in the licensees' commercial-grade dedication programs identified during recent inspections performed by the U.S. Nuclear Regulatory Commission (NRC). This generic letter also provides further discussion of the staff's views on key activities which, if included in licensee implementation of these programs, could have avoided such failures.

During 1986 to 1989, the NRC has conducted inspections of the licensees' procurement and commercial-grade dedication programs. During these inspections, the NRC staff identified a common, programmatic deficiency in the licensees' control of the procurement and dedication of commercial-grade items for safety-related applications. In a number of cases, the staff found that licensees had not maintained programs to ensure the suitability of equipment for safety-related applications. In addition, the staff identified equipment of indeterminate quality installed in the licensee's facilities.

The NRC staff believes that these inspection findings, in part, indicate a change in the industry's procurement practices and the decrease in the number of qualified nuclear-grade vendors. Ten years ago, licensees made most procurements for major assemblies from approved vendors with programs pursuant to Appendix B of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR). Currently, licensees are increasing the numbers of commercial-grade replacement parts that they procure for use in safety-related applications. This has resulted in a shift of responsibility for ensuring the quality of the item purchased from the suppliers to the licensees. Therefore, dedication processes for commercial-grade parts have increased in importance and NRC inspections have determined that a number of licensees have not satisfactorily performed this dedication process.

The industry should be fully aware of the NRC's concerns in this program area. In the past, escalated enforcement cases have provided notice to the affected licensees and to the industry of NRC's findings, concerns, and expectations in the implementation of procurement and dedication programs. Further, the NRC staff continues to participate in numerous industry meetings and conferences to discuss the NRC's positions in this area.

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The Nuclear Utility Management and Resources Council (NUMARC) Board of Directors recently approved a comprehensive procurement initiative which if effectively implemented should markedly improve the assurance that quality components are installed in nuclear power plants. While monitoring industry implementation of these programs, the NRC staff is deferring inspections of licensees' procurement and commercial-grade dedication processes for about a year to allow utilities to have sufficient time to fully understand and implement the guidance being developed by the industry.

However, the NRC will continue to perform certain types of inspection activities. For example, the staff will conduct selected assessments to determine the progress of the industry in improving the procurement and dedication processes. The staff will continue to perform reactive inspections relating to operational events or to defective equipment and, as required, will continue to initiate resultant enforcement actions which will not be affected by the decision to defer programmatic inspections. In addition, the staff will continue to perform inspections of vendors. To further encourage timely and effective implementation of the NUMARC initiatives, the staff will not initiate enforcement action in cases of past programmatic violations that have been adequately corrected. In addition, the staff does not expect licensees to review all past procurements. However, if during current procurement activities, licensees identify shortcomings in the form, fit, or function of specific vendor products, or if failure experience or current information on supplier adequacy indicates that a component may not be suitable for service, corrective actions should include a look-back for all such installed and stored items. In performing these actions for both stored and installed items, licensees should follow the existing requirements for corrective and follow-up actions contained in Criterion XVI of 10 CFR Part 50, Appendix B. A licensee should determine programmatic root causes when actual deficiencies in several different vendor products are identified during current procurement activities and when these deficiencies lead to the replacement of installed or warehouse items as part of corrective action. In such cases, a further sampling of previously procured commercial-grade items may be warranted.

NRC Generic Letter 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," described the NRC's observations on good practices in procurement and provided the NRC's conditional endorsement of an industry standard (EPRI NP-5652) on methods of commercial-grade procurement and dedication. A number of inspection findings indicate that licensees have failed to include certain key activities as appropriate in the implementation of the dedication process. Enclosure 1 includes further discussion of the NRC staff's views on the successful implementation of licensees' programs for commercial-grade dedication. The commercial-grade dedication approaches discussed in Enclosure 1 do not constitute new NRC requirements or positions. We will continue to meet with the industry to ensure a common understanding of implementation issues in this area.

DRAFT

Although no response to this letter is required, if you have any questions regarding this matter, please contact the persons listed below.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Characteristics of Effective Commercial-Grade Procurement and Dedication Programs
2. List of Recently Issued Generic Letters

Technical Contact: Richard P. McIntyre, NRR
(301) 492-3215

DRAFT

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Enclosure 1

CHARACTERISTICS OF EFFECTIVE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS

Background

Appendix B to 10 CFR Part 50 contains the NRC's regulations for procurement quality assurance (QA) and quality control (QC) for products to be used in safety-related applications. In addition, the NRC has provided further guidance in Regulatory Guides 1.28, 1.33, and 1.123. These requirements and guides assure the suitability of equipment, including commercial-grade items for use in safety-related systems. Criterion III of Appendix B requires licensees to select and review for suitability of application materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems, and components. Criterion IV requires that procurement documents specify the applicable requirements necessary to ensure functional performance. Criterion VII requires licensees to assure that the following are sufficient to identify whether specification requirements for the purchased material and equipment have been met: source evaluation and selection, objective evidence of quality, inspection of the source, and examination of products upon delivery. The process used to satisfy these requirements when upgrading commercial-grade items for safety-related applications is commonly called "dedication." The process of ensuring compliance with 10 CFR Part 50, Appendix B, must include all those activities necessary to establish and confirm the quality and suitability of those items to be installed in safety-related applications. Some of the dedication activities may occur early in the procurement cycle, before the item is accepted from the manufacturer. (10 CFR Part 21 has a more restricted definition of commercial-grade item dedication related to responsibility for evaluation and reporting of defects.) Generic Letter (GL) 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," discussed commercial-grade dedication in terms of engineering involvement in the procurement process, product acceptance, and the dedication process as identified in the EPRI NP-5652 guidelines. This enclosure provides examples of specific failures by licensees to fully implement certain key activities for dedicating and ensuring the suitability of commercial-grade products for safety-related applications. Appropriate implementation of these key activities would have avoided the failures in procurement and commercial-grade dedication observed during past NRC inspections.

Inspection Observations and Findings

From 1986 to 1989, headquarters and regional personnel conducted 13 team inspections of licensees' procurement and dedication programs. These inspections have identified a common, broad programmatic deficiency in licensees' control over the procurement and dedication of commercial-grade items. In a number of cases, licensees have not maintained programs to ensure

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the suitability of equipment for use in safety-related applications as required by 10 CFR Part 50, Appendix B, Criterion III. From these 13 inspections, the staff identified 8 findings that were considered to be Severity Level III violations and 3 findings that were Severity Level IV violations. At one plant, the staff did not assign a severity level to individual violations. Instead, the staff considered the entire group to be a Severity Level III problem and used enforcement discretion, as provided under the shutdown policy, based on the licensee's corrective actions (see 10 CFR Part 2, Appendix C, Section V.G.2). Only one of the plants that were inspected did not receive violations in this program area.

In GL 89-02, the NRC has conditionally endorsed the dedication methods described in EPRI NP-5652 guidelines. The staff believes that licensees who implement these dedications methods, in accordance with the NRC's endorsement, can establish a basis for satisfying the existing requirements of Appendix B to 10 CFR Part 50 as these requirements apply to the dedication process for commercial-grade items. An effective commercial-grade dedication program should include provisions to demonstrate that a dedicated item is suitable for safety-related applications. For a licensee to adequately establish suitability, certain key activities must be performed as appropriate as part of the dedication process.

During each of the 13 inspections, the staff identified a common element in each of the inspection findings. This element was the failure of the licensee to assure that a commercially procured and dedicated item was suitable for the intended safety-related application. In its ability to perform its intended safety function, a dedicated commercial-grade item should be equivalent to the same item procured under a 10 CFR Part 50, Appendix B QA program. The following is a list of the 13 licensees inspected and the inspection report numbers. A summary of the general inspection findings and NRC observations on these findings follows the list of licensee inspections.

LICENSEE and PLANT	INSPECTION REPORT NO.
1. Tennessee Valley Authority (Sequoyah)	50-327/86-61 50-328/86-61
2. Southern California Edison (San Onofre)	50-206/87-02 50-361/87-03 50-362/87-04
3. Alabama Power (Farley)	50-348/87-11 50-364/87-11
4. Louisiana Power and Light (Waterford)	50-382/87-19

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LICENSEE and PLANT		INSPECTION REPORT NO.
5.	Sacramento Municipal Utility District (Rancho Seco)	50-312/88-02
6.	Maine Yankee Atomic Power (Maine Yankee)	50-309/88-200
7.	Northern States Power (Prairie Island)	50-282/88-201 50-306/88-201
8.	Portland General Electric (Trojan)	50-344/88-39 50-344/88-46
9.	Connecticut Yankee Atomic Power (Haddam Neck)	50-213/89-200
10.	Washington Public Power Supply System (WNP-2)	50-397/89-21 50-397/89-28
11.	Florida Power (Crystal River)	50-302/89-200
12.	Gulf States Utilities (River Bend)	50-458/89-200
13.	Commonwealth Edison (Zion)	50-295/89-200 50-304/89-200

1. Inspection Findings

   a. Failure to identify the methods and acceptance criteria for verifying the critical characteristics, such as during receipt inspection, dedication process, or post-installation testing.

   b. Failure to establish verifiable, documented traceability of complex commercial-grade items to their original equipment manufacturers in those cases where the dedication program cannot verify the critical characteristics.

   c. Failure to recognize that some commercial-grade items cannot be fully dedicated once received on site. Certain items are manufactured using special processes, such as welding and heat treating. Dedication testing of these items as finished products would destroy them. For these items, licensees may need to conduct vendor surveillances or to witness certain activities during the manufacturing process.

Discussion

The NRC staff has met on several occasions with NUMARC and licensee representatives to discuss "critical characteristics" as used in the context of commercial-grade procurement and dedication. The term "critical

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characteristics" is not contained in Appendix B and has no special regulatory significance beyond its use and definition in various industry guides and standards. The NRC has not taken the position that all design requirements must be considered to be critical characteristics as defined and used in EPRI NP-5652. Rather, as stated in Appendix B, Criterion III, licensees must assure the suitability of all parts, materials, and services for their intended safety-related applications (i.e., there needs to be assurance that the item will perform its intended safety function when required). The licensee is responsible for identifying the important design, material, and performance characteristics for each part, material, and service intended for safety-related applications, establishing acceptance criteria, and providing reasonable assurance of the conformance of items to these criteria. There is no minimum or maximum number of critical characteristics that need to be verified. Further, the critical characteristics for an item may vary from application to application depending on the design and performance requirements unique to each application.

A licensee may take different approaches for the verification of the critical characteristics, depending on the complexity of the item. In many cases, the licensee can verify the critical characteristics of a simple item during the receipt inspection. However, for a complex item with internal parts which receive special processing during manufacturing, the licensee would probably need to audit or survey the vendor to verify the critical characteristics necessary for the item to perform its safety function. When the dedication program cannot verify the critical characteristics, the licensee should establish documented, verifiable traceability to the original equipment manufacturer. For simple items with critical characteristics that can be verified for the most severe or limiting plant application, the licensee might prefer a broad dedication program to identify and verify the item's critical characteristics to qualify that item for all possible plant applications. For complex items that would be purchased for specific plant applications, the licensee should address the acceptance criteria for each item individually. Engineering involvement is essential in either method because the technical evaluation will identify the critical characteristics, acceptance criteria, and the methods to be used for verification.

2. Inspection Findings

a. Failure to demonstrate that a like-for-like replacement item is identical in form, fit, and function to the item it is replacing. Part number verification is not sufficient because of the probability of undocumented changes in the design, material, or fabrication of commercial-grade items using the same part number.

b. Failure to evaluate changes in the design, material, or manufacturing process for the effect of these changes on safety function performance (particularly under design basis event conditions) of replacement items that are similar as opposed to identical to the items being replaced.

c. Failure to ensure that items will function under all design requirements. On some occasions, licensees only ensured that the commercial-grade item would function under normal operation conditions.

d. Failure to verify the validity of certificates of conformance received from vendors not on the licensee's list of approved vendors/suppliers. An unverified certificate of conformance from a commercial-grade vendor is not sufficient.

Discussion

A like-for-like replacement is defined as the replacement of an item with an item that is identical. A like-for-like replacement does not change the engineering analysis or as-built configuration of the component or system in which it is installed, and the replacement item meets the same design specifications, technical and quality requirements, and functional characteristics as the item it replaces. If differences from the original item are identified in the replacement item, then the item is not identical, but similar to the item being replaced, and an evaluation must be performed to determine if any changes in design, material, or the manufacturing process could impact the functional characteristics and ultimately the component's ability to perform its required safety function.

If the licensee can demonstrate that the replacement item is identical, then the licensee need not identify the safety function or review and verify the design requirements and critical characteristics. For example, the replacement item would be identical if it was purchased at the same time from the same vendor as the item it is replacing, or if the user can verify that there have been no changes in the design, materials, or manufacturing process since procurement of the item being replaced.

Engineering involvement is essential in the above activities. The extent of this involvement is dependent on the nature, complexity, and use of the items to be dedicated. Engineering personnel should participate in the procurement process, and product acceptance, to develop purchase specifications, determine specific testing requirements applicable to the products, and evaluate the test results. When engineering personnel specify design requirements for inclusion on the purchase documents for replacement components, they need not reconstruct and reverify the design adequacy, but only ensure that these design requirements (which may reference the original design basis) are properly translated into the purchase order.

Reliance on part number verification and certification documentation is insufficient to ensure the quality of commercially procured products. To conduct effective product acceptance programs, licensees should ensure that these programs include receipt and source inspection, appropriate testing criteria, effective vendor audits (including witness/hold points), special tests and inspections, and post-installation tests. The licensees should establish procedures to implement their programs and should ensure that the implementing personnel have adequate qualifications and training.

MEMORANDUM FOR: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

NOV 28 1990

FROM: Frank J. Miraglia, Deputy Director
Office of Nuclear Reactor Regulation

SUBJECT: PROPOSED GENERIC LETTER ON LICENSEE COMMERCIAL-GRADE PROCUREMENT AND DEDICATION PROGRAMS

The Office of Nuclear Reactor Regulation requests that the Committee to Review Generic Requirements (CRGR) consider the enclosed proposed generic letter. The staff is proposing the enclosed generic letter to notify the industry of the staff's intent to pause in conducting programmatic procurement inspection and enforcement activities and to identify a number of failures in the licensees' commercial-grade dedication programs identified during past NRC inspections. This generic letter also provides information from the NRC's inspections of the licensees' commercial-grade procurement and dedication programs which, if included in licensees' implementation of these programs, could have avoided violations of regulatory requirements.

The commercial-grade dedication inspection findings discussed in Enclosure 1 to the generic letter are based upon 10 CFR Part 50, Appendix B requirements and do not involve changes in the staff's positions. Further, the proposed generic letter does not require any specific licensee action or response to the NRC based on the issuance of this generic letter. Because no new regulations or regulatory practices are involved, the relation to the Commission's safety goals have not been explicitly addressed. However, this action appears to relate to how well a plant is operated. The matters addressed in this generic letter contribute to reducing or avoiding a substantial increase in uncertainty in the assumptions on which safety goal calculations are based.

Enclosure 2 to this memorandum is the proposed generic letter and Enclosure 1 contains the CRGR review package. Brian K. Grimes, Director, Division of Reactor Inspection and Safeguards, is the sponsoring division director. OGC concurrence is currently being sought.

Original signed by
Frank J. Miraglia
Frank J. Miraglia, Deputy Director
Office of Nuclear Reactor Regulation

OFC	:VIB:DRIS:NRR	:VIB:DRIS:NRR	:DD:DRIS:NRR	:D:DRIS:NRR	:	
NAME	:RPMcIntyre:mkm*	:UPotapovs*	:BDLiaw	:BKGrimes	:	
DATE	:10/25/90	:10/25/90	:11/ /90	:11/21/90	:	

OFC	:ADT:NRR	:DD:NRR	:Tech Editor			
NAME	:WTRussell	:FMiraglia	:JMain*			
DATE	:11/21/90	:11/28/90	:11/14/90			

OFFICIAL RECORD COPY DOCUMENT NAME: JORDAN MEMO

11/28/90

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Enclosure: see next page
1. CRGR Review Package
2. Draft generic letter on Licensee
   Commercial-Grade Procurement
   and Dedication Programs

CONTACT: Richard P. McIntyre, NRR
492-3215

* see previous concurrence

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REMARKS

This previous Central File material can now be made publicly available.

MATERIAL RELATED TO CAGR MEETING NO. 199

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MATERIAL RELATED TO CRGR MEETING NO. 199
TO BE MADE PUBLICLY AVAILABLE

1. MEMO FOR J. TAYLOR FROM E. JORDAN DATED 3-8-91
SUBJECT: MINUTES OF CRGR MEETING NUMBER 199
INCLUDING THE FOLLOWING ENCLOSURES WHICH WERE NOT
PREVIOUSLY RELEASED:

   a. ENCLOSURE 2
   A SUMMARY OF DISCUSSIONS OF A PROPOSED
   Draft final Amendment to 10 CFR 50, App. J
   on Containment Leakage Testing

   b. ENCLOSURE ____
   A SUMMARY OF DISCUSSIONS OF A PROPOSED

   c. ENCLOSURE ____
   A SUMMARY OF DISCUSSIONS OF A PROPOSED

2. MEMO FOR E. JORDAN FROM ____ DATED ____
FORWARDING REVIEW MATERIALS ON A PROPOSED

3. MEMO FOR E JORDAN FROM ____ DATED ____
FORWARDING REVIEW MATERIALS ON A PROPOSED

4. MEMO FOR E JORDAN FROM ____ DATED ____
FORWARDING REVIEW MATERIALS ON A PROPOSED

March 8, 1991

MEMORANDUM FOR: James M. Taylor
Executive Director for Operations

FROM: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

SUBJECT: MINUTES OF CRGR MEETING NUMBER 199

The Committee to Review Generic Requirements (CRGR) met on Tuesday, February 12, 1991 from 8:00 a.m. to 1:30 p.m. A list of attendees at the meeting is enclosed (Enclosure 1). The following items were discussed at the meeting:

1. The CRGR discussed a draft presentation outline for a Commission briefing on CRGR activities. The briefing was scheduled for February 22, 1991.

2. L. Shao and G. Arndt of RES continued a presentation (begun at Meeting No. 198) on a draft final amendment to 10 CFR 50, Appendix J on containment leakage testing. The CRGR recommended in favor of the amendment, subject to some revisions to be coordinated with the CRGR staff.

3. The CRGR briefly discussed planning and agenda matters. It was suggested that the next site visit be scheduled in June 1991.

Questions concerning these meeting minutes should be referred to Dennis Allison (492-4148).

Original Signed by:
E. L. Jordan

Edward L. Jordan, Chairman
Committee to Review Generic Requirements

Enclosures:
As stated

cc: Commission (5)
SECY
J. Lieberman
P. Norry
D. Williams
W. Parler
Regional Administrators
CRGR Members

Distribution:
Central File w/o encl.
PDR (NRC/CRGR) w/o encl.

P. Kadambi	CRGR C/F
J. Sniezek	M. Taylor
J. Heltemes	S. Treby
G. Arndt	L. Shao
D. Ross	J. Conran
D. Allison	W. Minners
A. Murphy	J. Costello

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DAA

CRGR:AEOD	DD:AEOD	C:CRGR:AEOD
DAllison:slm	DRoss	EJordan
03/6/91	03/1 /91	03/7/91

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ENCLOSURE 1

ATTENDANCE LIST

CRGR Meeting No. 199

February 12, 1991

CRGR Members

E. Jordan
G. Arlotto
J. Moore
F. Miraglia
B. Sheron

CRGR Staff

J. Conran
D. Allison

NRC Staff

L. Shao
G. Arndt
R. Bosnak
J. Pulsipher
J. Costello
A. Murphy
P. Kadambi
F. Mora

Enclosure 2 to the Minutes of CRGR Meeting No. 199
Draft Final Amendment to 10 CFR 50, Appendix J
on Containment Leakage Testing

February 12, 1991

TOPIC

L. Shao and G. Arndt of RES continued a presentation on a draft final amendment to 10 CFR 50, Appendix J on containment leakage testing. A proposed amendment had been published for comment in 1986. After considerable work and discussion with industry, this draft final amendment was being proposed.

The CRGR had been previously briefed on the draft final amendment at Meeting No. 192 and had begun its review at Meeting No. 198.

BACKGROUND

The review package was as described in the minutes of Meeting No. 198.

CONCLUSIONS/RECOMMENDATIONS

The CRGR recommended in favor of the amendment, subject to several revisions to be coordinated with the CRGR staff.

The action was considered to be a backfit. In the backfit analysis the staff concluded that there would be a substantial safety enhancement from better, more uniform tests and test reports, greater confidence in the reliability of the test results, fewer exemption requests and interpretive debates, withdrawing NRC endorsement of a superseded national standard, greater flexibility, and a refocusing of corrective actions to where problems originate. This increase in safety would be achieved without any overall change in net costs to the industry.

Although the CRGR believed that there would be a safety benefit from the action, it did not agree that this amendment would provide "... a substantial increase in the overall protection of the public health and safety ..." as required by the backfit rule. Nevertheless, the CRGR considered the amendment very worthwhile and strongly recommended forwarding it to the Commission.

The CRGR believed that the staff's consideration of safety goals was adequate.

Some of the additional comments/revisions are summarized below:

1. The package should explain the basis for the increase in radiation dose (10,000 person-rem). It should note that testing practices have evolved since 1985 and the differential (or additional dose) should be less when compared to current practice than it was in 1985 when the dose estimates were made. Doses shuld be discussed in the backfit analysis summary section in page 18.

2. Where exceptions are discussed, a model similar to 10 CFR 50.55a (a) (3) should be used.

3. The visual examination requirement, which is in the current rule but not the proposed rule, should be added as recommended by the staff.

4. The conclusion of the backfit analysis, on page 28, should be revised.

5. The definitions and usage of terms such as "primary containment" should be made consistent with other sections of the rules as well as regulatory guides, etc.

6. It should be made more clear that the first LLRT is to be done at the first refueling outage and subsequent tests are to be performed at alternate outages.

7. Delete the last part of the last sentence of page 42.

8. Reconsider the maximum time between Type A tests. Thirty months may be appropriate rather than 26 months.

9. Reconsider use of terms such as "leakage," "leak rate," and "improve leakage."

10. Remove the condition "for which no identical ... plants" on page 52.

11. Remove the section on Multiple Leakage Barriers or Subatmospheric Containments on page 53.

12. Ensure reporting of failed LLRT's are covered under 10 CFR 50.73, and remove from Appendix J on page 55.

13. In their submittals, licensees should identify the exemptions to the current Appendix J which should remain in effect under the new Appendix J. The submittal requirement should call for a schedule rather than a plan.

14. In the regulatory guide.

    (a) Remove "thereby reducing an unproductive ..." on page 3.

    (b) Indicate "The staff always applies the single active failure criterion to the review of containment isolation systems in item 4 on page 4.

    (c) Reconsider and revise as appropriate the statement in 6.1(12.1) on page 9.

    (d) Reconsider and revise as appropriate the implementation section.

March 8, 1991

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CRGR 5

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DAllison:slm	DRoss	EJordan
03/6/91	03/1/91	03/7/91

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4pp.

ENCLOSURE 1

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CRGR Meeting No. 199

February 12, 1991

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Draft Final Amendment to 10 CFR 50, Appendix J
on Containment Leakage Testing

February 12, 1991

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    (b) Indicate "The staff always applies the single active failure criterion to the review of containment isolation systems in item 4 on page 4.

    (c) Reconsider and revise as appropriate the statement in 6.1(12.1) on page 9.

    (d) Reconsider and revise as appropriate the implementation section.

May 16, 1994

Docket No. 50-346

Mr. Donald C. Shelton
Senior Vice President, Nuclear
Centerior Service Company
c/o Toledo Edison Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, Ohio 43449

Dear Mr. Shelton:

SUBJECT: CORRECTION TO AMENDMENT NOS. 181, 182, 183 AND 184 TO FACILITY OPERATING LICENSE NO. NPF-3 (TAC NOS. M86933, M84912, M87339 AND M86426)

On November 19, 1993, December 16, 1993, December 30, 1993, and December 30, 1993, the Commission issued in respective order, Amendment Nos. 181, 182, 183, and 184. The amendments revised the Technical Specifications in response to your applications dated June 23, 1993, November 9, 1992, August 30, 1993, and May 6, 1993, respectively.

Technical Specification pages transmitted with the subject amendments contained typographical errors as follows: page 5-5 (Amendment No. 181), page B 3/4 5-2 (Amendment No. 182), page B 3/4 6-4 (Amendment No. 183), page 1-6a and page 3/4 4-12 (Amendment No. 184). Those errors have been corrected and the corrected pages are enclosed.

Please accept our apologies for any inconvenience these administrative errors may have caused you.

Sincerely,
ORIGINAL SIGNED BY

Garmon West, Jr., Acting Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosure:
TS pages 5-5, B 3/4 5-2, B 3/4 6-4, 1-6a and 3/4 4-12

cc w/enclosure:
See next page

OFFICE	LA:PDIII-3	(A)PM:PDIII-3	PM:PDIII-3	PD:PDIII-3
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 16, 1994

Docket No. 50-346

Mr. Donald C. Shelton
Senior Vice President, Nuclear
Centerior Service Company
c/o Toledo Edison Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, Ohio 43449

Dear Mr. Shelton:

SUBJECT: CORRECTION TO AMENDMENT NOS. 181, 182, 183 AND 184 TO FACILITY OPERATING LICENSE NO. NPF-3 (TAC NOS. M86933, M84912, M87339 AND M86426)

On November 19, 1993, December 16, 1993, December 30, 1993, and December 30, 1993, the Commission issued in respective order, Amendment Nos. 181, 182, 183, and 184. The amendments revised the Technical Specifications in response to your applications dated June 23, 1993, November 9, 1992, August 30, 1993, and May 6, 1993, respectively.

Technical Specification pages transmitted with the subject amendments contained typographical errors as follows: page 5-5 (Amendment No. 181), page B 3/4 5-2 (Amendment No. 182), page B 3/4 6-4 (Amendment No. 183), page 1-6a and page 3/4 4-12 (Amendment No. 184). Those errors have been corrected and the corrected pages are enclosed.

Please accept our apologies for any inconvenience these administrative errors may have caused you.

Sincerely,

Garmon West, Jr.

Garmon West, Jr., Acting Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosure:
TS pages 5-5, B 3/4 5-2, B 3/4 6-4,
1-6a and 3/4 4-12

cc w/enclosure:
See next page

Mr. Donald C. Shelton
Toledo Edison Company

Davis-Besse Nuclear Power Station
Unit No. 1

cc:

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Ohio Environmental Protection Agency
DERR--Compliance Unit
ATTN: Zack A. Clayton
P. O. Box 1049
Columbus, Ohio 43266-0149

State of Ohio
Public Utilities Commission
180 East Broad Street
Columbus, Ohio 43266-0573

Mr. James R. Williams
State Liaison to the NRC
Adjutant General's Department
Office of Emergency Management
Agency
2825 West Granville Road
Columbus, Ohio 43235-2712

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,110 ± 200 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 1% delta k/k for calculation uncertainty.

b. A rectangular array of stainless steel cells spaced 12 31/32 inches on centers in one direction and 13 3/16 inches on centers in the other direction. Fuel assemblies stored in the spent fuel pool shall be placed in a stainless steel cell of 0.125 inches nominal thickness or in a failed fuel container.

c. Fuel assemblies stored in the spent fuel pool in accordance with Technical Specification 3.9.13.

5.6.1.2 The new fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 1% delta k/k for uncertainties as described in Section 9.1 of the USAR.

b. A K_{eff} equivalent to less than or equal to 0.98 when immersed in a hydrogenous "mist" of such a density that provides optimum moderation (i.e., highest value of K_{eff}), which includes a conservative allowance of 1% delta k/k for uncertainties as described in Section 9.1 of the USAR.

c. A nominal 21 inch center-to-center distance between fuel assemblies placed in the storage racks.

d. Fuel assemblies having a maximum initial enrichment of 5.0 weight percent uranium-235.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below 9 feet above the top of the fuel storage racks.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 735 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limit of Table 5.7-1.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 CORE FLOODING TANKS

The OPERABILITY of each core flooding tank ensures that a sufficient volume of borated water will be immediately forced into the reactor vessel in the event the RCS pressure falls below the pressure of the tanks. This initial surge of water into the vessel provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on volume, boron concentration and pressure ensure that the assumptions used for core flooding tank injection in the safety analysis are met.

The tank power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with a core flooding tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one tank is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems with RCS average temperature $\geq$ 280°F ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the core flooding tanks is capable of supplying sufficient core cooling to maintain the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 280°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. The decay heat removal system leak rate surveillance requirements assure that the leakage rates assumed for the system during the recirculation phase of the low pressure injection will not be exceeded.

Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

Containment Emergency Sump Recirculation Valves DH-9A and DH-9B are de-energized during MODES 1, 2, 3 and 4 to preclude postulated inadvertent opening of the valves in the event of a Control Room fire, which could result in draining the Borated Water Storage Tank to the Containment Emergency Sump and the loss of this water source for normal plant shutdown. Re-energization of DH-9A and DH-9B is permitted on an intermittent basis during MODES 1, 2, 3 and 4 under administrative controls. Station procedures identify the precautions which must be taken when re-energizing these valves under such controls.

Borated Water Storage Tank (BWST) outlet isolation valves DH-7A and DH-7B are de-energized during MODES 1, 2, 3, and 4 to preclude postulated inadvertent closure of the valves in the event of a fire, which could result in a loss of the availability of the BWST. Re-energization of valves DH-7A and DH-7B is permitted on an intermittent basis during MODES 1, 2, 3, and 4 under administrative controls. Station procedures identify the precautions which must be taken when re-energizing these valves under such controls.

3/4.5.4 BORATED WATER STORAGE TANK

The OPERABILITY of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on BWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the BWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The bottom 4 inches of the borated water storage tank are not available, and the instrumentation is calibrated to reflect the available volume. The limits on water volume, and boron concentration ensure a pH value of between 7.0 and 11.0 of the solution sprayed within the containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components.

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leakage rate are consistent with the assumptions used in the safety analyses. The leak rate surveillance requirements assure that the leakage assumed for the system during the recirculation phase will not be exceeded.

Borated Water Storage Tank (BWST) outlet isolation valves DH-7A and DH-7B are de-energized during MODES 1, 2, 3, and 4 to preclude postulated inadvertent closure of the valves in the event of a fire, which could result in a loss of the availability of the BWST. Re-energization of valves DH-7A and DH-7B is permitted on an intermittent basis during MODES 1, 2, 3 and 4 under administrative controls. Station procedures identify the precautions which must be taken when re-energizing these valves under such controls.

Containment Emergency Sump Recirculation Valves DH-9A and DH-9B are de-energized during MODES 1, 2, 3, and 4 to preclude postulated inadvertent opening of the valves in the event of a fire, which could result in draining the Borated Water Storage Tank to the Containment Emergency Sump and the loss of this water source for normal plant shutdown. Re-energization of valves DH-9A and DH-9B is permitted on an intermittent basis during MODES 1, 2, 3, and 4 under administrative controls. Station procedures identify the precautions which must be taken when re-energizing these valves under such controls.

3/4.6.2.2 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the required time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. Containment isolation valves and their required isolation times are addressed in the USAR. The opening of a closed inoperable containment isolation valve on an intermittent basis during plant operation is permitted under administrative control. Operating procedures identify those valves which may be opened under administrative control as well as the safety precautions which must be taken when opening valves under such controls.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the Hydrogen Analyzers, Containment Hydrogen Dilution System, and Hydrogen Purge System ensures that this equipment will be available to maintain the maximum hydrogen concentration within the containment vessel at or below three volume percent following a LOCA.

The two redundant Hydrogen Analyzers determine the content of hydrogen within the containment vessel. The Hydrogen Analyzers, although they have their OPERABILITY requirements in this Specification, are considered part of the post-accident monitoring instrumentation of Specification 3/4.3.3.6, Post-Accident Monitoring Instrumentation.

The Containment Hydrogen Dilution (CHD) System consists of two full capacity, redundant, rotary, positive displacement type blowers to supply air to the containment. The CHD System controls the hydrogen concentration by the addition of air to the containment vessel, resulting in a pressurization of the containment and suppression of the hydrogen volume fraction.

The Containment Hydrogen Purge System Filter Unit functions in conjunction with the CHD System and is designed to release air from the containment atmosphere through a HEPA filter and charcoal filter prior to discharge to the station vent.

As a backup to the CHD System and the Containment Hydrogen Purge System, the capability to install an external hydrogen recombination system has been provided.

3/4.6.5 SHIELD BUILDING

3/4.6.5.1 EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the emergency ventilation systems ensures that containment vessel leakage occurring during LOCA conditions into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. This requirement is necessary to meet the assumptions used in the safety analyses and limit the site boundary radiation doses to within the limits of 10 CFR 100 during LOCA conditions.

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1.29 Deleted

PROCESS CONTROL PROGRAM

1.30 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

1.31 Deleted

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.32 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Specifications 6.9.1.10 and 6.9.1.11.

1.33 Deleted

1.34 Deleted

1.35 Deleted

1.36 Deleted

MEMBER(S) OF THE PUBLIC

1.37 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

SITE BOUNDARY

1.38 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

UNRESTRICTED AREA

1.39 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes. The definition of UNRESTRICTED AREA used in implementing the Radiological Effluent Technical Specifications has been expanded over that in 10 CFR 20.3(a)(17). The UNRESTRICTED AREA boundary may coincide with the exclusion (fenced) area boundary, as defined in 10 CFR 100.3(a), but the UNRESTRICTED AREA does not include areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the LIMITING CONDITIONS FOR OPERATION to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a.

1.40 Deleted

CORE OPERATING LIMITS REPORT

1.41 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7. Plant operation within these core operating limits is addressed in individual specifications.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

TABLE 4.4-2
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G. (1)	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair by sleeving defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug or repair by sleeving defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair by sleeving defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug or repair by sleeving defective tubes and inspect 2S tubes in each other S.G. Report to the NRC prior to resumption of plant operation.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or repair by sleeving defective tubes. Report to the NRC prior to resumption of plant operation.	N/A	N/A

(1) $S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

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CHAIRMAN

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

May 13, 1994

The Honorable Michael O. Leavitt
Governor of Utah
Salt Lake City, Utah 84114-0601

Dear Governor Leavitt:

Thank you for your letter of May 2, 1994, requesting that the NRC modify the license of Umetco Minerals for its White Mesa uranium mill facility near Blanding, Utah. Your request is being treated as a Petition filed pursuant to 10 CFR 2.206. The Petition has been referred to the Director, Office of Nuclear Materials, Safety and Safeguards, for prompt evaluation and action. The NRC will respond to the issues you raised as soon as your request is evaluated.

I have enclosed a copy of the notice that is being filed with the Office of the Federal Register for publication.

Sincerely,

Ivan Selin

Enclosure:
Federal Register Notice

DF03 1/1

CHAIRMAN

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

May 13, 1994

The Honorable Arnold Christensen
President of the Utah Senate
Salt Lake City, Utah 84114

Dear Mr. Christensen:

Thank you for your letter of May 2, 1994, requesting that the NRC modify the license of Umetco Minerals for its White Mesa uranium mill facility near Blanding, Utah. Your request is being treated as a Petition filed pursuant to 10 CFR 2.206. The Petition has been referred to the Director, Office of Nuclear Materials, Safety and Safeguards, for prompt evaluation and action. The NRC will respond to the issues you raised as soon as your request is evaluated.

I have enclosed a copy of the notice that is being filed with the Office of the Federal Register for publication.

Sincerely,

Ivan Selin

Enclosure:
Federal Register Notice

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

CHAIRMAN

May 13, 1994

The Honorable Rob W. Bishop
Speaker of the Utah House
of Representatives
Salt Lake City, Utah 84114

Dear Mr. Bishop:

Thank you for your letter of May 2, 1994, requesting that the NRC modify the license of Umetco Minerals for its White Mesa uranium mill facility near Blanding, Utah. Your request is being treated as a Petition filed pursuant to 10 CFR 2.206. The Petition has been referred to the Director, Office of Nuclear Materials, Safety and Safeguards, for prompt evaluation and action. The NRC will respond to the issues you raised as soon as your request is evaluated.

I have enclosed a copy of the notice that is being filed with the Office of the Federal Register for publication.

Sincerely,

Ivan Selin

Enclosure:
Federal Register Notice

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NO. ____

UMETCO MINERALS

LICENSE NO. _____

RECEIPT OF PETITION FOR DIRECTOR'S DECISION UNDER 10 CFR 2.206

Notice is hereby given that by Petition dated May 2, 1994, Michael O. Leavitt, Governor of the State of Utah, and the Utah Legislature (Petitioners) requested that the U.S. Nuclear Regulatory Commission (NRC) take action regarding Umetco Minerals, licensee of the White Mesa uranium mill facility near Blanding, Utah. Petitioners request that the NRC modify the Umetco Minerals license to reflect the licensee's original request for authority to receive 5,000 cubic yards of in-situ waste at the White Mesa facility, rather than the 10,000 cubic yards authorized by a license amendment granted by the NRC on June 2, 1993. Petitioners also request that the NRC confer with the State of Utah prior to future license amendments and receive the concurrence of the Utah Legislature and Governor before approving future license amendments involving mill tailing disposal in Utah.

Petitioners assert as the basis for this request that the NRC has in effect created the equivalent of a commercial waste disposal facility for in-situ mining waste unlicensed by Utah, while ignoring Utah's waste policy and laws.

The Petition has been referred to the Director of the Office of Nuclear Materials, Safety and Safeguards pursuant to 10 CFR 2.206. As provided by Section 2.206, appropriate action will be taken on the specific issues raised by the Petition in a reasonable time.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert M. Bernero, Director
Office of Nuclear Materials, Safety
and Safeguards

Dated at Rockville, Maryland
this __ th day of May 1994

STATE OF UTAH

OFFICE OF THE GOVERNOR
SALT LAKE CITY
84114-0601

MICHAEL O. LEAVITT
GOVERNOR

OLENE S. WALKER
LIEUTENANT GOVERNOR

May 2, 1994

Chairman Ivan Selin
US Nuclear Regulatory Commission
Washington DC 20555

Dear Chairman Selin:

This is a request to 10 CFR 2.206 regarding a licensing action by the NRC Uranium Recovery Field Office (URFO). On June 2, 1993, NRC URFO issued the Umetco White Mesa Mill in Blanding, Utah, a license amendment with a condition which allows Umetco to receive in-situ mining waste for disposal. Umetco's request was for 5,000 cubic yards of in-situ waste. NRC URFO approved the amendment with a general license condition which allows Umetco White Mesa Mill to receive 10,000 cubic yards of in-situ waste annually for disposal from any single source. While NRC may have been attempting to apply consistent national conditions to all uranium mills in the United States, we believe NRC in effect has created the equivalent of a commercial waste disposal facility for in-situ mining waste in Utah.

During the 1994 Utah legislative session, the Umetco license amendment raised issues of concern which resulted in the passage of Senate Concurrent Resolution 11 (enclosed). As indicated by the Resolution, Utah policy-makers feel strongly about waste issues within our borders. Utah law requires waste facilities desiring to operate on a commercial basis to meet siting criteria and obtain gubernatorial and legislative approval before they can operate. The NRC approval is perceived to ignore Utah's waste policy and laws. We would request that NRC reconsider the license amendment issued to Umetco Minerals and modify the amendment to reflect the original request of 5,000 cubic yards.

Chairman Ivan Selin
Page 2
May 2, 1994

We appreciate the opportunity to bring this important waste policy issue to your attention.

Sincerely,

Arnold Christensen
President of the Senate

Rob W. Bishop
Speaker of the House

Michael O. Leavitt
Governor

Enclosure
MOL/DRN/doc

RESOLUTION REGARDING NRC ACTION REGARDING

DISPOSAL OF URANIUM BY-PRODUCT

1994

GENERAL SESSION

Enrolled Copy

S. C. R. No. 11

By Stephen J. Rees

Craig A. Peterson

David H. Steele

Blaze D. Wharton

Haven J. Barlow

Arnold Christensen

A CONCURRENT RESOLUTION OF THE LEGISLATURE AND THE GOVERNOR EXPRESSING CONCERN OVER A DECISION BY THE NUCLEAR REGULATORY COMMISSION TO AMEND A URANIUM MILL SITE LICENSE TO CREATE A COMMERCIAL RADIOACTIVE WASTE FACILITY THAT IS UNLICENSED BY THE STATE OF UTAH; URGING THE NUCLEAR REGULATORY COMMISSION TO MODIFY THE LICENSE AMENDMENT, TO CONFER WITH THE STATE OF UTAH PRIOR TO FUTURE LICENSE AMENDMENTS, AND TO RECEIVE THE CONCURRENCE OF THE LEGISLATURE AND GOVERNOR BEFORE APPROVING FUTURE LICENSE AMENDMENTS INVOLVING MILL TAILING DISPOSAL IN UTAH; AND URGING THE DEPARTMENT OF ENVIRONMENTAL QUALITY TO VERIFY SAFETY RESTRICTIONS ARE IN PLACE AT THE WHITE MESA FACILITY.

Be it resolved by the Legislature of the state of Utah, the Governor concurring therein:

WHEREAS on August 2, 1993, the staff of the Region IV offices of the Nuclear Regulatory Commission authorized the amendment of a license issued to a uranium mill site facility at White Mesa, near Blanding, Utah to allow the facility to recycle for the purposes of recovering of uranium content and to receive and dispose of waste by-product materials generated from off-site licensed uranium leaching facilities;

WHEREAS the facility will be engaged in receipt of waste material and processing for recovery of uranium content;

WHEREAS the amendment to the White Mesa Facility license was issued without public notice, without the opportunity for public comment, and without adequate environmental review;

WHEREAS the state of Utah has established by statute, policies, and procedures regarding the number of waste disposal facilities, the amounts of out-of-state wastes to be allowed to be received by Utah, and the types of waste that can be disposed in Utah;

WHEREAS the initial license amendment request for the White Mesa Facility was for disposal of 5,000 cubic yards of waste;

WHEREAS the Nuclear Regulatory Commission granted a license amendment which allows for disposal of a maximum of 10,000 cubic yards of uranium mill tailings annually from each individual source; and

WHEREAS the Nuclear Regulatory Commission license amendment has in effect created the equivalent of a commercial waste disposal facility for uranium mill tailings:

NOW, THEREFORE, BE IT RESOLVED that the Legislature of the state of Utah, the Governor concurring therein, express deep concern over the Nuclear Regulatory Commission's action in authorizing the White Mesa Facility to receive uranium mill tailings in excess of the amount requested in the license amendment application and from facilities other than requested in the application.

BE IT FURTHER RESOLVED that the Legislature and the Governor urge the operator of the White Mesa Facility to prepare and submit to the Utah Department of Environmental Quality detailed information which includes source of material, analytical information, and economic analysis to demonstrate the economic viability for reprocessing and recycling of each waste stream.

BE IT FURTHER RESOLVED that the Legislature and the Governor urge that the Nuclear Regulatory Commission modify the license amendment to restrict waste disposal to that originally requested.

BE IT FURTHER RESOLVED that the Legislature and the Governor urge the Nuclear Regulatory Commission to confer with the state of Utah and provide opportunity for comment prior to the issuance of license amendments involving uranium mill tailings disposal in Utah.

BE IT FURTHER RESOLVED that the Legislature and the Governor urge the Utah Department of Environmental Quality to ensure that appropriate and applicable restrictions and controls necessary to protect public health and the environment are in place at the White Mesa Facility.

BE IT FURTHER RESOLVED that the Legislature and the Governor urge the Nuclear Regulatory Commission to not issue a license amendment in Utah involving disposal of uranium mill tailings from a facility without receiving the concurrence of the Legislature and the Governor of the state of Utah.

BE IT FURTHER RESOLVED that copies of this resolution be sent to the Nuclear Regulatory Commission, the Utah Department of Environmental Quality, and Utah's congressional delegation.